CEN-386-NP-A

Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/kgU for Combustion Engineering 16x16 PWR Fuel

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Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/kgU for Combustion Engineering 16x16 PWR Fuel

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LEGAL NOTICE

CEN-386-NP-A

Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/kgU for Combustion Engineering 16x16 PWR Fuel

August 1992

ABB Combustion Engineering Nuclear Fuel

ABB

ASEA BROWN BOVERI

Generic Approval of Topical Report CEN-386-P
"Verification of the Acceptability of a 1-Pin Burnup
Limit of 60 MWD/kgU for Combustion Engineering
16x16 PWR Fuel"

dated
June 22, 1992

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

June 22, 1992

Mr. A. E. Scherer, Director
Nuclear Licensing
Combustion Engineering, Inc.
P. O. Box 500
Windsor, Connecticut 06095

Dear Mr. Scherer:

SUBJECT: GENERIC APPROVAL OF C-E TOPICAL REPORT CEN-386-P, "VERIFICATION OF THE ACCEPTABILITY OF A 1-PIN BURNUP LIMIT OF 60 MWD/kg FOR COMBUSTION ENGINEERING 16X16 PWR FUEL (TAC NO. M82192)

On November 14, 1991, you requested NRC review and generic approval of the C-E topical report CEN-386-P, entitled "Verification of The Acceptability of A 1-Pin Burnup Limit of 60 MWD/kg for Combustion Engineering 16X16 PWR Fuel." The methodology described in the topical report CEN-386-P was approved for licensing applications for ANO-2 and St. Lucie 2 in NRC safety evaluations dated November 27, 1990, and October 18, 1991, respectively. Based on your submittal and review of the previously approved SERs, we conclude that CEN-386-P is not necessarily plant-specific for ANO-2 or St. Lucie 2, and therefore CEN-386-P can be applied generically to other C-E 16x16 plants. The NRC staff was supported in this review by our consultant, the Pacific Northwest Laboratory, who previously provided input to the approval for applications to ANO-2 and St. Lucie 2. In summary, the NRC staff approves the generic applicability of CEN-386-P for licensing applications. Our evaluation applies only to matters described in the topical report.

In accordance with procedures established in NUREG-0390, "Topical Report Review Status," we request that C-E publish accepted versions of this topical report, proprietary and non-proprietary, within 3 months of receiving this letter. The accepted versions shall include an "A" (designating accepted) following the report identification symbol, and shall include this letter and the ANO-2 SER dated November 27, 1990.

If our criteria or regulations change such that we can no longer accept this report, applicants referencing this topical report will be expected to revise and resubmit their respective documentation, or submit justification that the topical report continues to apply without revision of their respective documentation.

Sincerely,

A C Thadani

Ashok C. Thadani, Director
Division of Systems Technology
Office of Nuclear Reactor Regulation

Enclosure:
ANO-2 Safety Evaluation

Request for
Generic Approval of Topical Report CEN-386-P
"Verification of the Acceptability of a 1-Pin Burnup
Limit of 60 MWD/kgU for Combustion Engineering
16x16 PWR Fuel"

ASEA BROWN BOVERI

November 14, 1991
LD-91-057

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington D.C. 20555

Subject: Submittal of Topical Report CEN-386-P, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWd/kg for Combustion Engineering 16x16 PWR Fuel"

References:

1) Letter, S. R. Peterson (NRC) to N. S. Carns (Entergy Operations), "Issuance of Amendment No. 111 to Facility Operating License No. NPF-6 - Arkansas Nuclear One, Unit No. 2 (TAC No. 74139)," November 27, 1990.

2) Letter, J. A. Norris (NRC) to J. H. Goldberg (FP&L), "St. Lucie Unit 2 - Fuel Pin Burnup to 60 MWd/kg," October 18, 1991.

3) CENPD-269-P, Revision 1-P, "Extended Burnup Operation of Combustion Engineering PWR Fuel," July 1984.

Gentlemen:

This letter submits (as Enclosure I) twenty-three (23) copies of Combustion Engineering's (C-E's) Topical Report CEN-386-P, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWd/kg for Combustion Engineering 16x16 PWR Fuel." C-E requests that the NRC review this Topical Report for generic application to C-E 16x16 PWR fuel.

The NRC has already provided plant-specific approval of Topical Report CEN-386-P for application in Arkansas Nuclear One Unit No. 2 and St. Lucie Unit 2 (References 1 and 2, respectively). Recently, the NRC was requested to approve Topical Report CEN-386-P for Waterford Unit 3. In view of the generic nature of the Topical Report, the NRC requested that

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C-E submit Topical Report CEN-386-P for generic approval. Generic approval will preclude the need for the NRC staff to review similar information on individual plant dockets and will allow utilities to plan for its implementation on their plant.

Topical Report CEN-386-P describes C-E's methodology for the evaluation of C-E 16x16 PWR fuel for 1-pin burnup up to 60 MWd/kg. The methodology extends and modifies previously approved C-E methodology for the evaluation of C-E PWR fuel for 1-pin burnups up to 52 MWd/kg, described in Reference 3.

As requested by Mr. L. E. Phillips (NRC) in a telephone conversation with Mr. Mario Robles (C-E) on October 3, 1991, C-E has identified limitations to the generic applicability of the Topical Report. These limitations are discussed in Enclosure II.

The material contained in Enclosure I is considered by C-E to be proprietary. As such, it is requested that the material be withheld from public disclosure in accordance with the provisions of 10CFR2.790 and that this material be appropriately safeguarded. The reasons for the classification of this material as proprietary are delineated in the affidavit provided as Enclosure III.

Correspondence regarding the review fee payments for this report should be directed to Combustion Engineering. If there are any questions on the Topical Report, or if we can be of other assistance to you or your Staff on this subject during the review process, please do not hesitate to call either me at (203) 285-5213 or Mr. Mario Robles of my staff at (203) 285-5215.

Very truly yours,

COMBUSTION ENGINEERING, INC.

S. A. Toelle

S. A. Toelle
Manager
Operating Reactor Licensing

SAT:lw

Enclosures: As Stated
(Enclosure I - Copies 000039 to 000061)

cc: L. E. Phillips (NRC)

Enclosure I to
LD-91-057

COMBUSTION ENGINEERING TOPICAL REPORT CEN-386-P

VERIFICATION OF THE ACCEPTABILITY OF A 1-PIN BURN-UP LIMIT OF 60 MWd/kg FOR COMBUSTION ENGINEERING PWR FUEL

JUNE 1989

Limitations to Generic Applicability of CEN-386-P for 16x16 PWR Fuel

Introduction

Topical Report CEN-386-P, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWd/kg for Combustion Engineering 16x16 PWR Fuel,"[1] describes C-E's methodology for the evaluation of C-E 16x16 PWR fuel for 1-pin burnups up to 60 MWd/kg. The methodology either extends or modifies previously approved C-E methodology to the evaluation of C-E PWR fuel for 1-pin burnups up to 52 MWd/kg[2].

Since Arkansas Nuclear One Unit No. 2 (ANO-2) had the lead high fuel burnup C-E 16x16 PWR fuel, CEN-386-P was submitted for NRC review on the ANO-2 docket. Following a supplemental submittal of responses to NRC review questions[3], the NRC approved CEN-386-P for ANO-2. The NRC subsequently authorized an amendment to the ANO-2 operating license that increased the rod-average fuel burnup limit to 60 MWd/kg[4].

The purpose of this enclosure is to identify ary limitations to the generic applicability of the CEN-386-P methodology for other C-E 16x16 PWR fuel. Observations made by the NRC in their Safety Evaluation Report[5] (SER) resulting from the ANO-2 review are also identified. Finally, for completeness, references made by the NRC in the SER to other C-E methods that were reviewed and approved on the ANO-2 docket are also provided since generic approval of CEN-386-P would require that equivalent approved methodology be used for all plants for which CEN-386-P will be applicable.

Discussion

Limitations Contained in Topical Report

CEN-386-P contains two specific limitations, or conditions, that are pertinent to the generic application of the methodology for extending the fuel burnup limit to 60 MWd/kg. These conditions are as follows:

> In Section 4.2.2.1.a of CEN-386-P, the methodology in the Topical Report is described to be applicable to certain fuel assembly guide tube properties that are proprietary to C-E. These properties must, therefore, be present for the methodology to be applicable.

In Section 4.1.2.2.2.a of CEN-386-P, it was noted that factors affecting cladding corrosion performance at extended burnup cannot differ substantially from that in the existing database without conducting additional corrosion evaluations. The specific factors identified in CEN-386-P are:

- average linear heat rate
- reactor coolant temperature
- reactor coolant lithium level

Furthermore, CEN-386-P states that the impact of cladding changes to improve in-reactor corrosion resistance compared to the cladding used in the current database should be included in these additional evaluations.

Observations Cited in ANO-2 Safety Evaluation Report

The SER cites two specific observations that may be construed as limitations with respect to the generic application of the methodology for extending the fuel burnup limit to 60 MWd/kg. These observations are as follows:

In Section 2.0(a) of the SER, it was observed that the stress analyses at extended burnup should include the effects of cladding thinning due to cladding oxidation. C-E agrees with this observation. In Section 4.1.2.2.a of CEN-386-P, however, the cladding wastage associated with the upper bound on oxide thickness was observed to be negligible with respect to cladding stress evaluations. If in the future cladding wastage is found not to be negligible, then C-E will explicitly include this effect in the cladding stress analyses. This observation does that impose a limitation on the generic applicability of the methodology in CEN-386-P.

In Section 2.0(e) of the SER, it was further observed that since cladding oxidation is dependent upon reactor-specific conditions (e.g., reactor coolant temperature and water chemistry), it is necessary to examine cladding oxidation on a reactor-specific basis. C-E also agrees with this observation, as it is wholly consistent with the second limitation contained in the Topical Report which was discussed above.

References to Other Methodology

In addition to the observations above, the SER also referred

to other C-E methodology that was reviewed by the NRC and approved on the ANO-2 docket. These other approved methodologies were part of the basis for approval of CEN-386-P. Presumably, the generic approval of CEN-386-P would similarly require that equivalent methodology be approved and used for all the plants to which CEN-386-P would be applicable. The C-E methodologies that were reviewed and approved on the ANO-2 docket and which were referenced in the SER are contained in the following:

> CEN-214(A)-P, "CETOP-D Code Structure and Modelling Methods for Arkansas Nuclear One - Unit 2," July 1982.
>
> CEN-289(A)-P, "Revised Rod Bow Penalties for Arkansas Nuclear One Unit 2," December 1984.

For C-E PWRs other than ANO-2, approved CETOP-D methodology is used that is equivalent to that described in CEN-214(A)-P. For C-E PWRs other than ANO-2, the approved rod bow methodology described in CENPD-225-P-A, "Fuel & Poison Rod Bowing," June 1983, is used in place of that described in CEN-289(A)-P.

Conclusion

The discussion above identifies that the only limitations to the generic applicability of the CEN-386-P methodology for other C-E 16x16 PWR fuel are described within CEN-386-P. Thus, it is concluded that the NRC can provide generic approval of the CEN-386-P methodology for all C-E 16x16 PWR fuel.

References

1. CEN-386-P, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWd/kg for Combustion Engineering 16x16 PWR Fuel," June 1989.

2. CENPD-269-P, Revision 1-P, "Extended Burnup Operation of Combustion Engineering PWR Fuel," July 1984.

3. CEN-386-P, Supplement 1-P, "Responses to Questions on Combustion Engineering Report CEN-386-P, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWd/kg for Combustion Engineering 16x16 PWR Fuel," April 1990.

4. Letter from S. R. Peterson (USNRC) to N. S. Carns (Entergy Operations, Inc.), "Issuance of Amendment No. 111 to Facility Operating License No. NPF-6 - Arkansas Nuclear One, Unit No. 2 (TAC No. 74139)," November 27, 1990.

5. "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 111 to Facility Operating License No. NPF-6, Entergy Operations, Inc., Arkansas Nuclear One, Unit No. 2, Docket No. 50-386," November 27, 1990.

Safety Evaluation Report (SER) of
Topical Report CEN-386-P
"Verification of the Acceptability of a 1-Pin Burnup
Limit of 60 MWD/kgU for Combustion Engineering
16x16 PWR Fuel"

on Arkansas Unit 2
dated
November 27, 1990

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 111 TO

FACILITY OPERATING LICENSE NO. NPF-6

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT NO. 2

DOCKET NO. 50-368

A. INTRODUCTION

Pursuant to 10 CFR 50.90 and 50.91, Entergy Operations, Inc. (the licensee) proposes to amend Facility Operating License No. NPF-6 for Arkansas Nuclear One, Unit No. 2 (ANO-2) in its submittal dated November 21, 1990. The amendment would authorize operation of ANO-2 with rod-average fuel burnups up to 60 megawatt days per kilogram of uranium (MWd/kgM) and is based on a Combustion Engineering report as discussed below.

B. EVALUATION

1.0 Discussion

On July 20, 1989, the Arkansas Power and Light Company requested the U.S. Nuclear Regulatory Commission (NRC) to review the Combustion Engineering, Inc. (CE) report CEN-386-P to support Arkansas Nuclear One, Unit 2 (ANO-2) operation with rod-average fuel burnups up to 60 megawatt days per kilogram of uranium (MWd/kgM) for CE 16x16 fuel. The analysis used to demonstrate that the fuel design criteria are met are presented in References 1 and 2. It should be noted that Reference 2 is a topical report previously approved by NRC (Reference 3) that extended the burnup level of CE designed fuel to 52 MWd/kgM (rod-average). The difference between References 1 and 2 is the incremental increase in rod-average burnup from 52 to 60 MWd/kgM for the CE 16x16 fuel design.

Presented in this report is a review of the CE mechanical design criteria, analysis methods, and results for the ANO-2 fuel design application for CE 16x16 fuel. This review was conducted to assure that when the design criteria/limits are met they will prevent fuel damage or failure and maintain fuel coolability, as defined in the Standard Review Plan (SRP) (Reference 4) up to rod-average burnups of 60 MWd/kgM.

This review was based on the licensing requirements identified in Section 4.2 of the SRP (Reference 4). The objectives of this fuel system safety review, as described in Section 4.2 of the SRP, are to provide assurance that 1) the fuel system is not damaged as a result of normal operation and anticipated operational

occurrences (AOOs), 2) the number of fuel rod failures is not underestimated for postulated accidents, 3) fuel system damage is never so severe as to prevent control rod insertion when it is required, and 4) coolability is always maintained. A "not damaged" fuel system is defined as one wherein fuel rods do not fail, fuel system dimensions remain within operation tolerances, and functional capabilities are not reduced below those assumed in the safety analyses. Objective 1 above is consistent with General Design Criterion (GDC) 10 (10 CFR Part 50, Appendix A) (Reference 5), and the design limits that accomplish this are called specified acceptable fuel design limits (SAFDLs).

"Fuel rod failure" (Objective 2) 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 (Reference 6) for postulated accidents. The general requirements to maintain control rod insertability (Objective 3) and core coolability (Objective 4) appear repeatedly in the GDC (e.g., GDC 27 and 35). Specific coolability requirements for the loss of coolant accident (LOCA) are given in 10 CFR 50.46 (Reference 7). "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 even after a severe accident.

In order to assure that the above stated objectives are met and follow the format of Section 4.2 of the SRP, this review covers the following three major categories: 1) Fuel System Damage Mechanisms, which are most applicable to normal operation and AOOs, 2) Fuel Rod Failure Mechanisms, which apply to normal operation, AOOs, and postulated accidents, and 3) Fuel Coolability, which is applied to postulated accidents. Specific fuel damage or failure mechanisms are identified under each of these categories in Section 4.2 of the SRP and these individual mechanisms are addressed in this report. The design criteria, analysis methods, and results for the 16x16 fuel design, up to a rod-average burnup of 60 MWd/kgM, will be discussed in this report under each fuel damage or failure mechanism.

Pacific Northwest Laboratory (PNL) has acted as a consultant to the NRC in this review. As a result of the review of the subject topical report by the NRC staff and their PNL consultants, a list of questions was sent by the NRC to the licensee (Reference 8) requesting further justification on why low measured cladding ductilities, greater cladding oxidation, guide wear, cladding collapse, and axial assembly growth are not limiting at the burnup level requested. The licensee has provided responses to these questions in References 9 and 10. The design criteria and analyses submitted for ANO-2 in support of this license submittal are those defined in CE reports (References 1 and 2) and, therefore, will be referred to as CE design criteria and analyses. The responses submitted by ANO-2 in this review were jointly developed by ANO-2 and CE staff and, therefore, will be referred to as ANO-2/CE responses.

The CE 16x16 design description is provided in Reference 11. The fuel damage and failure mechanisms and CE analyses of these mechanisms are addressed in Sections 2.0 and 3.0, respectively, while fuel coolability is addressed in Section 4.0.

2.0 FUEL SYSTEM DAMAGE

The design criteria presented in this section should not be exceeded during normal operation, including AOOs. Under each damage mechanism, there is an evaluation of the design criteria analysis methods and analyses used by CE to demonstrate that fuel damage does not occur for the 16x16 design during normal operation, including AOOs up to a rod-average burnup of 60 MWd/kgM.

(a) Stress

Bases/Criteria - In keeping with the GDC 10 SAFDLs, fuel damage criteria for stress should ensure that fuel system dimensions remain within operational tolerances for normal operation and AOOs, and that functional capabilities are not reduced below those assumed in the safety analysis. The CE 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 (Reference 2).

The CE stress criteria for the fuel assembly components are provided in References 11 and 12. The design limit for fuel rod and burnable poison rod cladding is that the maximum primary tensile stress is less than two-thirds of the Zircaloy yield strength as affected by temperature.

The design limit of the Inconel X-750 upper-end fitting spring is that the calculated shear stress will be less than or equal to the minimum yield stress in shear.

Many of these bases and limits are used by the industry at large. CE has employed various conservatisms in the limits such as the use of unirradiated yield strengths for zirconium-based alloys. The NRC has concluded (Reference 3) that the fuel assembly, fuel rod, burnable poison rod, and upper-end fitting spring stress design bases and limits were acceptable for rod-average burnup levels up to 52 MWd/kgM. Extending the burnup level to 60 MWd/kgM does not reduce the applicability of these criteria, and thus, these criteria are found acceptable for use in the current ANO-2 applications for the CE 16x16 design.

Evaluation - CE has stated that the methods used to perform stress analyses will not change from those used and approved for previous applications. These analyses are performed using conventional engineering formulas from standard engineering mechanics textbooks and performed in accordance with ASME general guidelines for analyzing primary and secondary stresses. The NRC has concluded (Reference 3) that these stress analyses are acceptable for rod-average burnup levels up to 52 MWd/kgM. Extending the rod-average burnup level to 60 MWd/kgM does not reduce the applicability of these methods and thus these analysis methods are found to be acceptable for application to the CE 16x16 design up to a rod-average burnup of 60 MWd/kgM. As noted in Section 3.0(e), stress analyses at extended burnup levels must include the effects of cladding thinning due to cladding oxidation.

(b) Design Strain

Bases/Criteria - With regard to fuel assembly design strain, the CE design basis for normal operation and AOOs is that permanent fuel assembly deflections shall not result in control element assembly (CEA) insertion time beyond that allowable. This basis is satisfied by adherence to the stress criteria mentioned above and strain criterion yet to be discussed.

The submitted topical report provides a design criterion for fuel rod and burnable poison rod cladding uniform circumferential strain (elastic plus plastic) of one percent (1%) as a means of precluding excessive cladding deformation. This strain criterion is consistent with that given in Section 4.2 of the SRP.

The material property that could have a significant impact on the cladding strain criterion at the requested extended burnup levels is cladding ductility. The strain criterion could be impacted if cladding ductility were decreased, as a result of extended burnup operations, to a level that would allow cladding failure without the 1% cladding strain criterion being exceeded in the CE analyses.

Recent measured cladding and plastic cladding strain values from CE fuel rods (Reference 13) and other pressurized water reactor (PWR) fuel vendors (Reference 14) have shown a decrease in cladding ductilities when local burnups exceed 52 MWd/kgM. The cladding plastic strain values decreased to 0.03 from 0.11% when local burnups were between 55 and 63 MWd/kgM. ANO-2/CE was questioned on whether these significant reductions in cladding plastic ductilities justified a decrease in the 1% design criterion for total uniform strain (elastic plus lastic) for CE fuel with local burnups greater than 55 MWd/kgM (Reference 13).

ANO-2/CE has responded (Reference 9) that because of the increase in the yield strength and the corresponding increase in elastic strain of the cladding due to irradiation, the typical elastic strains were above 1% using nominal values for irradiated yield strength and Young's modulus at burnups greater than 55 MWd/kgM. ANO-2/CE was further questioned about the probability that the combined elastic plus plastic strains between 55 and 63 MWd/kgM would fall below the 1% strain criterion. ANO-2/CE presented (Reference 10) a statistical analysis of their measured yield strength data from cladding with local burnups greater than 55 MWd/kgM and calculated a two-sided tolerance limit about the mean value for yield strength. They also calculated a two-sided tolerance limit about the mean value for Young's modulus using data from the open literature. Using the lower bound tolerance limit for yield strength and the upper bound tolerance limit for Young's modulus, plus the range of plastic strain, they calculated that there is a 9% probability that cladding strain would fall below the 1% total limit for a strain limit.

This reviewer has performed an independent simplified statistical analysis at a 5% probability level that total uniform strain will fall below 1% using a one-sided lower tolerance limit of the measured yield strengths at burnups greater than 55 MWd/kgM and a one-sided upper tolerance limit of the measured values for Young's modulus. This analysis has demonstrated that there is slightly less than a 5% probability that cladding strain will fall below the 1% total uniform strain limit. The 5% probability of falling below the 1% strain limit calculated by this reviewer is conservative because this simplified approach has assumed that combining the yield strength and Young's modulus tolerance limits will result in an equivalent plastic strain tolerance limit. Hall and Sampson (Reference 15) have provided a more exact analytical procedure for determining either one-sided or two-sided tolerance limits for the distribution of the quotient (e.g., plastic strain) of two independent normal variables (e.g., yield strength and Young's modulus) for this application.

Therefore, because 1) there is a very low probability of total uniform strain falling below 1% in the CE 16x16 fuel cladding, 2) histories are used in the CE strain analysis, and 3) no fuel failures have been observed on fuel rods irradiated with rod-average burnups to 63 MWd/kgM, we conclude that the 1% total uniform strain limit remains applicable for the ANO-2 use of the CE 16x16 fuel design up to a rod-average burnup of 60 MWd/kgM. However, it should be stressed that future requests to extend the rod-average burnup limit beyond 60 MWd/kgM should be accompanied with measured cladding strain, yield, and fracture strength data at the extended burnup levels requested. This data is necessary to demonstrate that the total uniform strain criterion of 1% remains applicable at these higher burnups and that fuel cladding brittle fracture will not occur during normal operation and AOOs at these higher burnups.

Evaluation - CE utilizes the FATES3B (Reference 16) computer code to predict cladding strain and other fuel performance phenomena at high burnup levels. This code has been approved by the NRC for fuel performance analyses up to rod-average burnups of 60 MWd/kgM (Reference 17). The FATES3B code will take the place of the earlier FATES3 code (Reference 18). The use of the FATES3B code for calculating cladding strain is acceptable for rod-average burnups up to 60 MWd/kgM.

(c) Strain Fatigue

Bases/Criteria - The ANO-2/CE strain fatigue criterion is different from those described in Section 4.2 of the SRP, viz., a safety factor of 2 on stress amplitude or of 20 on the number of cycles using the methods of O'Donnell and Langer (Reference 19). Instead, CE has proposed in the past that the cumulative strain cycling usage (i.e., the sum of the ratios of the number of cycles in a given effective strain range to the permitted number in that range) will not exceed 0.8. For Zircaloy cladding, the design limit curve has been adjusted to provide a strain margin for the

effects of uncertainty and irradiation. The resulting curve given in References 2 and 11 bounds all of the data used in the development of the criterion that is discussed in the SRP. The NRC has previously concluded that the proposed criterion was acceptable for current burnup levels (Reference 3).

The material property that could have a significant effect on the strain fatigue criterion is cladding ductility. As discussed in the above section for design strain, extended burnup operation above local burnups of 55 MWd/kgM has demonstrated a significant reduction in cladding ductilities. However, as also discussed herein, there is a low probability that cladding ductility will fall below the acceptable limit for total uniform strain at a rod-average burnup of 60 MWd/kgM. In addition, there is a considerable amount of conservatism in the ANO-2/CE strain fatigue calculation. Therefore, we conclude that the strain fatigue criterion proposed in Reference 1 is acceptable for licensing applications to CE 16x16 fuel up to a rod-average burnup of 60 MWd/kgM.

Evaluation - The fuel and cladding models used to determine fuel and cladding diametral strain for the fatigue analysis are those in the FATES3B code (Reference 16) which has been approved by the NRC (Reference 17). The power history used for the fatigue analysis includes conservative estimates of daily power cycling and AOOs and has been described previously in Reference 2. This analysis also accounts for a conservative number of hot and cold shutdowns during the fuel lifetime. This power history takes into account the extra duty required for rod-average burnups up to 60 MWd/kgM. Therefore, we conclude that the strain fatigue analysis models referenced are acceptable for application to the ANO-2 use of the CE 16x16 fuel design up to a rod-average burnup of 60 MWd/kgM.

(d) Fretting Wear

Bases/Criteria - Fretting wear is a concern for fuel and burnable poison rods and the guide tubes. Fretting 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 spacer spring loads in combination with small amplitude, flow-induced, vibratory forces. Guide tube wear may result when there is flow-induced vibration between the control rod ends and the inner wall of the guide tubes.

While Section 4.2 of the SRP 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 report has addressed fuel and burnable poison rod fretting wear by referring to Reference 2 and stating that no significant wear has been observed for CE fuel rods, and no additional fretting wear was expected due to the extension of rod-average burnup level to 60 MWd/kgM. Indicated in Reference 2 is that a specific fretting wear limit was not used for CE fuel assembly components, because it has not been a problem for current CE fuel designs. This same data was used to explain why fretting wear was not accounted for in the fuel and burnable poison rod analyses for cladding stress and fatigue. In order to support this claim, in the previous review, CE provided fuel examination information from 744 assemblies with average burnups up to approximately 52 MWd/kgM that showed no failures or significant wear on the surface of their fuel or burnable poison rods. It is noted that since this time, CE has performed a visual examination of 14x14 designed fuel rods irradiated to rod-average burnups up to 56 MWd/kgM and found no surface anomalies other than minor scratches (Reference 13).

Due to the lack of significant fretting wear in the examination of more than 744 CE fuel assemblies, with rod-average burnups to 56 MWd/kgM and existing fuel surveillance programs, we conclude that CE has demonstrated that fretting wear in their fuel and burnable poison rods will be acceptable up to rod-average burnups of 60 MWd/kgM.

Guide tube wear, however, was observed in several CE fuel assemblies in 1977. Since then a design change in the guide tubes has greatly reduced guide tube wear for both 14x14 and 16x16 fuel assembly designs. However, it was noted in the NRC review of Reference 2 that very limited low burnup data were available for this new guide tube design (Reference 3). For this submittal, ANO-2/CE was requested (Reference 8) to provide guide tube wear data for the new unsleeved guide tube design to be used in the subject reload and future CE 16x16 plant reloads and compare this data to their maximum predicted wear correlation. ANO-2/CE has provided (Reference 9) this comparison, which demonstrates that the measured wear data is a factor of 3 below the CE correlation for maximum wear. However, it should be noted that the maximum in-reactor operating times of the wear data are only one-third of those expected for rod-average burnups to 60 MWd/kgM. The ANO-2/CE response has argued that this lack of wear data at the maximum burnup level requested is satisfactory because 1) the CE maximum guide tube fretting wear correlation is very conservative, and 2) there is a large margin between maximum predicted fretting wear at the maximum burnup level requested and the minimum amount of allowable wear that a guide tube can sustain without violating any design criteria.

Due to the conservative nature of the CE guide tube fretting wear correlation and the large margin that exists before design criteria are violated, we conclude that guide tube wear in the CE 16x16 fuel design is acceptable up to a rod-average burnup level of 60 MWd/kgM.

Evaluation - The ANO-2/CE submittal has suggested that the lack of a large amount of measured fretting wear in CE fuel and burnable poison rods supports their conclusion that they do not need to include the effects of cladding thinning due to fretting wear in their stress, strain, and fatigue analyses for the fuel and burnable poison rods. However, this does not answer the question of what effect the calculated impact of a small reduction in cladding thickness has on safety and design analyses, e.g., LOCA and stress/strain. In the past, CE (Reference 2) has indicated that the most limiting LOCA analysis is early-in-life when stored energy is the highest and fretting wear is insignificant for this analysis. We agree with this assessment. ANO-2/CE has also responded to a question on cladding thinning due to oxidation by conservatively reducing cladding thickness of the 16x16 fuel rods by 3 mils in their stress analysis [see Section 3.0(e)]. This inclusion of cladding thinning due to corrosion is judged to bound thinning due to fretting wear because corrosion is the greater of the two thinning mechanisms and because these two mechanisms do not occur simultaneously at the same location on a fuel rod. For example, where fretting wear is present on the fuel or burnable poison rod, oxidation will not be present and vice versa. Therefore, it is concluded that cladding thinning of the fuel and burnable poison rods due to fretting wear are bounded by CE's analysis of cladding thinning due to oxidation.

As noted in the "Criteria" section, guide tube wear has been a problem in the past for CE assemblies. Design changes have been implemented by CE for both 14x14 and 16x16 assemblies to reduce guide tube wear. Both out-of-reactor and in-reactor confirmation tests have been performed to show that these design changes have resulted in a significant decrease in guide tube wear for in-reactor residence times that are one-third of those expected for an extended burnup level of 60 MWd/kgM. Extrapolating the guide tube wear to the in-reactor residence time expected for an extended rod average burnup level of 60 MWd/kgM has demonstrated that guide tube wear will remain at a relatively low level. We conclude that guide tube wear is not expected to be a problem up to a rod-average burnup of 60 MWd/kgM for the newly designed guide tubes in the CE 16x16 design (based on the low level of wear at lower burnups). The licensee should continue to examine guide tubes up to the extended burnup levels requested to confirm that wear is not a problem at these burnup levels.

(e) <u>Oxidation and Crud Buildup</u>

Bases/Criteria - Section 4.2 of the SRP identifies cladding oxidation and crud buildup as potential fuel system damage mechanisms. General mechanical properties of the cladding are not significantly impacted by thin oxides or crud buildup. The major means of controlling fuel damage due to cladding oxidation and crud is through water chemistry controls, materials used in the primary system, and fuel surveillance programs that are all reactor specific. Because these controls are already included in the specific reactor design, a design limit on cladding oxidation and crud is considered to be redundant, and thus, not necessary.

This does not, however, eliminate the need to include the effects of cladding oxidation and crud in safety analyses such as for LOCA and mechanical analyses. This will be discussed in further detail in the evaluation presented below.

Evaluation - As noted above, the amount of cladding oxidation expected for a particular reactor is dependent on fuel rod powers (surface heat flux), chemistry controls, and primary inlet coolant temperatures used by that reactor, but the amount of oxidation increases with in-reactor residence time and cannot be eliminated. Therefore, extending the rod-average burnup level to 60 MWd/kgM could result in 1) thicker oxide layers that provide an extra thermal barrier that increases cladding and fuel temperatures, and 2) cladding thinning that can affect the mechanical analyses. The degree of this effect on thermal and mechanical analyses is dependent on reactor coolant temperatures and the level of success of a reactor's chemistry controls.

The ANO-2/CE submittal (Reference 1) has provided oxide thickness measurements from fuel rod cladding irradiated in ANO-2 near the burnup level requested and placed a conservative upper bound limit on the measured values. The upper bound oxide thickness at a rod-average burnup of 60 MWd/kgM was used to estimate the increase in cladding temperatures and stress, and found to have little impact on either of these analyses. Therefore, we conclude that cladding oxidation is acceptable for the CE 16x16 fuel design in ANO-2 up to a rod-average burnup of 60 MWd/kgM.

There is an indication that cladding corrosion may limit the fuel rod performance lifetime for higher burnup irradiations for specific plants. Because cladding oxidation is dependent on reactor-specific conditions, such as reactor coolant temperatures and water chemistry, it is necessary to examine cladding oxidation on a reactor-specific basis. Also, future requests to extend the rod-average burnup limit beyond 60 MWd/kgM should be accompanied with reactor-specific corrosion data at the burnup levels requested.

(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 transfer to the coolant. Rather than placing design limits on the amount of bowing that is permitted, the effects of bowing are included in the safety analysis. This is consistent with the SRP and the NRC has approved this for current burnup levels (Reference 3). The methods used for predicting the degree of rod bowing at the extended burnups requested are evaluated below.

Evaluation - The CE analysis methods used to account for the effect of fuel and poison rod bowing in 14x14 and 16x16 fuel assemblies are presented

in Reference 2 and CENPD-225 (Reference 20) with its supplements. These methods have been approved by the NRC (References 3 and 20) for fuel and Type 3 poison rods to current burnup levels.

Reference 2 has compared 14x14 rod bow data with burnups to 45 MWd/kgM to their licensing rod bow model and demonstrated that the model becomes more conservative at higher burnups. These data appeared to suggest that the rate of rod bow significantly decreases at burnups greater than 30 to 35 MWd/kgM while the CE analytical model for rod bow assumes little or no decrease in the rate of rod bowing with burnup. This results in very conservative predictions of rod bowing in CE 14x14 designed fuel at high burnup levels. Reference 2 has also demonstrated that the CE rod bowing model for 16x16 fuel rods was very conservative by comparison to data with burnups up to 33 MWd/kgM. ANO-2 has indicated that they routinely perform visual examination of their fuel assemblies to provide assurances of satisfactory performance of their fuel. The phenomenon of rod bowing is generic to all LWRs even though design differences such as the length between spacers and rod diameter are important to the amount of rod bowing. Therefore, other fuel vendor experience with rod bowing is valuable in evaluating the trend in rod bowing at extended burnups.

FRAMATOME has measured rod bow on their FRAGEMA fuel assemblies for fuel burnups up to 53 MWd/kgM and found that the rate of rod bowing versus burnup decreases at burnups greater than 30 to 35 MWd/kgM (Reference 21). Similar measurements of rod bowing have been made by Kraftwerk Union AG (KWU) on their fuel designs up to burnups of 50 MWd/kgM (Reference 22) and found that due to the scatter in their limited data, the decrease in the rate of rod bowing was not as evident as that demonstrated in References 2 and 21. However, KWU did find that rod bowing was limited to gap closures of less than 40% on their fuel designs which is consistent with the data in Reference 2.

We conclude that the CE analysis methods (Reference 20) applied to the CE 16x16 fuel design in ANO-2 will remain conservative up to the extended burnup level requested and, therefore, are acceptable up to a rod-average burnup level of 60 MWd/kgM.

(g) Axial Growth

Bases/Criteria - The core components requiring axial-dimensional evaluation are the CEAs, burnable poison rods, fuel rods, and fuel assemblies. The CEAs are not included in this extended burnup review. The growth of burnable poison and fuel rods is mainly governed by a) the irradiation and stress-induced growth of the Zircaloy-4 cladding, and b) the behavior of poison, fuel, and spacer pellets, and their interaction with the Zircaloy-4 cladding. The growth of the fuel assemblies is a function of both the comprehensive creep and the irradiation-induced growth of the Zircaloy-4 guide tubes. For the Zircaloy cladding and fuel assembly guide tubes,

the critical tolerances that require controlling are a) the spacing between the fuel rods and the upper fuel assembly fitting (i.e., shoulder gap), and b) the spacing between the fuel assemblies and the core internals. Failure to adequately design for the former may result in fuel rod bowing, and for the latter, may result in collapse and failure of the assembly hold-down springs. With regard to inadequately designed shoulder gaps, problems have been reported (References 23, 24, 25, and 26) in foreign (Obrigheim and Beznau) and domestic (Ginna and ANO-2) plants that have necessitated predischarge modifications to fuel assemblies.

For burnable poison and fuel rods, CE has a design basis that sufficient shoulder gap clearances must be maintained throughout the design lifetime of the fuel at a 95% confidence level. Similarly, for fuel assembly axial growth, CE has a design basis that sufficient clearance must be maintained between the fuel assembly and the upper guide structure throughout the design lifetime of the fuel assembly at a 95% confidence level. This basis allocates a fuel assembly gap spacing which will accommodate the maximum axial growth, when establishing the design minimum initial fuel assembly clearance with respect to the core internals. These design bases and limits dealing with axial growth prevent mechanical interference and thus have been approved by NRC for previous extended burnup levels (Reference 3). We conclude that these design bases and limits will ensure that contact is prevented, and thus, are found to be acceptable for the CE 16x16 fuel design to 60 MWd/kgM.

Evaluation - The CE methods and models used for predicting fuel rod and assembly growth in this submittal (Reference 1) have been changed somewhat from those previously approved to better predict the new higher exposure growth data. This evaluation will discuss the new revised models used to predict fuel rod and assembly growth. We will then discuss how CE uses these revised models to predict 1) the shoulder gap spacings between the fuel rod and the upper fuel assembly fitting, and 2) the gap spacing between the fuel assembly and core internals.

The new revised fuel and burnable poison rod growth model is based on CE 14x14 and 16x16 rod data with rod-average burnups above those requested. The model predicts a "best estimate" value of rod growth with uncertainties. The new revised assembly growth model is based on the SIGREEP computer code and growth data from assemblies with stress relief annealed (SRA) guide tubes with assembly average burnups below those requested in this submittal. The SIGREEP prediction of assembly growth takes into account the different axial stresses on the guide tubes for different CE plant fuel assemblies including the ANO-2 assemblies and uses input parameters with assigned statistical uncertainties along with Monte Carlo random selection techniques and combinations of these uncertainties to obtain a probability density function of assembly growth at a given fluence (burnup) level.

The CE evaluation of shoulder gap spacing uses the lower bound probability density function for assembly growth and the upper bound probability density function for rod growth with uncertainties in the SIGREEP computer code to predict the shoulder gap at an upper bound 95% probability with a 95% confidence level. This CE methodology for predicting an upper bound 95/95 shoulder gap spacing has been compared to measured shoulder gap data (Reference 1) that have assembly-average burnups below those requested in this submittal. These CE upper bound predictions do indeed bound the shoulder gap data and appear to become even more conservative at the higher burnup levels. It should be noted that in the shoulder gap calculation the amount of fuel rod growth is much greater than the amount of assembly growth, therefore, the prediction of fuel rod growth dominates the analysis of shoulder gap spacing. It should also be noted that the CE rod growth data have rod-average burnups greater than those requested in this submittal.

We conclude that the CE analysis methodology is acceptable for application to the CE 16x16 design up to a rod-average burnup of 60 MWd/kgM because 1) CE has fuel rod growth data above the burnup level requested, 2) fuel rod growth dominates the shoulder gap spacing analysis, and 3) the large amount of conservative margin CE has demonstrated in their prediction of shoulder gap spacing.

The CE analysis of the gap spacing between the upper fuel assembly and core internals uses the SIGREEP probability density function for assembly growth to predict a minimum 95/95 value for this gap spacing in order to prevent bottoming out of the assembly hold-down springs. Because CE does not have assembly growth data up to the burnup level requested, they were questioned (Reference 8) on the gap margin that exists at the burnup level requested in this submittal to prevent bottoming of the hold-down spring. ANO-2/CE's response (Reference 9) indicated that there was approximately one-third of the original as-fabricated gap spacing left prior to bottoming out of the hold-down spring at the burnup requested. Due to this significant margin and CE's conservative analysis methodology, we conclude that bottoming out and failure of the hold-down spring due to fuel assembly growth is not expected for the CE 16x16 design up to a rod-average burnup of 60 MWd/kgM. However, we encourage ANO-2 to visually examine the hold-down springs for those assemblies discharged with rod-average burnups near or at the 60 MWd/kgM level.

(h) Rod Internal Pressure

Bases/Criteria - Rod internal pressure is a driving force for, rather than a direct mechanism of, fuel system damage that could contribute to the loss of dimensional stability and cladding integrity. Section 4.2 of the SRP presents a rod pressure limit that is sufficient to preclude fuel damage in this regard, and it has been widely used by the industry; it states that rod internal gas pressure should remain below the nominal system pressure during normal operation, unless otherwise justified. CE

has elected to justify a rod internal pressure limit above system pressure in Reference 27 and this proprietary rod pressure limit has been approved by the NRC.

The CE design criterion used to establish this proprietary rod pressure limit is: "The fuel rod internal hot gas pressure shall not exceed the critical maximum pressure determined to cause an outward cladding creep rate that is in excess of the fuel radial growth rate anywhere locally along the entire active length of the fuel rod." In addition, CE has evaluated the impact of this rod pressure limit on hydride reorientation and accident analyses. The NRC approved rod pressure limit defined in Reference 27 is also acceptable for application to the CE 16x16 fuel design to a rod-average burnup of 60 MWd/kgM.

Evaluation - CE has indicated that they will use the FATES3B (Reference 16) computer code to calculate maximum rod internal pressures and this code has been approved by NRC in Reference 17. The FATES3B code has been verified against fission gas release data from a variety of fuel designs with rod-average burnups up to 60 MWd/kgM. The use of the approved FATES3B code is recommended over the earlier approved FATES3A code (Reference 18) because the former has been verified against a much larger data base at higher burnup levels.

ANO-2/CE were questioned on the apparent small underprediction of fission gas release by the FATES3B code when fission gas release values were low (<3% release) at high burnup levels and the impact of this underprediction on licensing analyses. ANO-2/CE responded that licensing analyses are typically performed in a conservative manner on the peak operating rod, i.e., a rod with high temperatures, high fission gas release and high internal rod pressures, and therefore, the small underprediction in fission gas release at low temperatures were insignificant for licensing analyses. They also demonstrated that the amount of underprediction was small in terms of calculated internal rod pressures in these low temperature rods. We concur with this assessment and conclude that the FATES3B code is acceptable for the analysis of internal rod pressures for the CE 16x16 fuel design up to a rod-average burnup of 60 MWd/kgM.

In addition to the computer code, the input power history to the code is very important for the internal rod pressure calculation. Consequently, CE has been required by NRC in the past, to define a methodology for determining the power history for the rod pressure calculation. This methodology was first reviewed and approved for Reference 2 and CE has provided an example of how this methodology is applied in Reference 1. We conclude that the use of the approved FATES3B code along with the approved CE power history methodology described in References 1 and 2 is acceptable for licensing applications for the CE 16x16 fuel design to a rod-average burnup of 60 MWd/kgM.

(i) Assembly Liftoff

Bases/Criteria - The SRP calls for the fuel assembly hold-down capability (wet weight and spring forces) to exceed worst-case hydraulic loads for normal operation, which includes AOOs. The NRC-approved CE Extended Burnup Topical Report (Reference 2) has endorsed this design basis. This is also found to be acceptable for application to the CE 16x16 fuel design up to a rod-average burnup of 60 MWd/kgM.

Evaluation - CE methodology for assembly liftoff analysis has been summarized in Reference 2 and approved by the NRC for current burnups in Reference 3. The fuel assembly liftoff force is a function of plant coolant flow, spring forces, and assembly dimensional changes. Extended burnup irradiation will result in additional hold-down spring relaxation and assembly length increases which will have opposing effects on the assembly hold-down force, i.e., the length increase will compress the spring, and therefore, increase the hold-down force. Industry experience has demonstrated that the assembly length increase due to irradiation more than compensates for spring relaxation so that the hold-down force increases with increased burnup. In fact, a major concern at extended burnups is that the assembly length change will compress the spring to the extent that it will bottom out and break. This issue has been addressed satisfactorily in Section 3.0(g), "Axial Growth." Consequently, we conclude that the issue of assembly liftoff has been satisfactorily addressed for the CE 16x16 fuel design to a rod-average burnup of 60 MWd/kgM.

(j) Control Material Leaching

Bases/Criteria - The SRP and GDC require that reactivity control be maintained. Rod reactivity can sometimes be lost by leaching of certain poison materials if the cladding of control-bearing material has been breached.

Evaluation - Reactivity loss from burnable poison rods at extended burnup levels is found to be insignificant because nearly all of the reactivity controlling boron-10 is burned out at these burnup levels. Consequently, reactivity loss due to leaching of burnable poison rods at the extended burnup level requested is considered to be insignificant.

Control rod lifetimes are not changed in this submittal from those previously approved by the NRC, and therefore, are not affected by this request to extend fuel rod-average burnups up to 60 MWd/kgM. We conclude that the issue of control material leaching has been satisfactorily addressed for the CE 16x16 fuel design up to a rod-average burnup of 60 MWd/kgM.

3.0 FUEL ROD FAILURE

In the following paragraphs, fuel rod failure thresholds and analysis methods for the failure mechanisms listed in the SRP are reviewed. When the failure

thresholds are applied to normal operation including AOOs, they are used as limits (and hence SAFDLs) since fuel failure under those conditions should not occur according to the traditional conservative interpretation of GDC 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 Part 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 are reviewed below.

(a) Hydriding

Bases/Criteria - Internal hydriding as a cladding failure mechanism is precluded by controlling the level of hydrogen impurities during fabrication. The moisture level in the uranium dioxide fuel is limited by CE to a proprietary value less than 20 ppm, and this specification is compatible with the ASTM specification (Reference 28) which allows two micrograms of hydrogen per gram of uranium (i.e., 2 ppm). This is the same as the limit described in the SRP and has been found acceptable by NRC (Reference 3) and continues to be acceptable for application to the CE 16x16 fuel design up to a rod-average burnup of 60 MWd/kgM.

External hydriding due to waterside corrosion is a possible reason for the observed ductility decrease at local burnups >55 MWd/kgM discussed in Section 2.0(b). Garde (Reference 29) has recently proposed that the ductility decrease is due to a combination of hydride formation and irradiation damage at these high burnup levels. The issue of cladding ductility has already been discussed in Section 2.0(b) and found to be acceptable for the CE 16x16 design to a rod-average burnup of 60 MWd/kgM.

Evaluation - The issue of internal hydriding is not expected to be affected by an increase in rod-average burnup level because this failure mechanism is dependent on the amount of hydrogen impurities introduced during fuel fabrication. Fuel failures due to internal hydriding occur early in a fuel rods lifetime and are not dependent on the length of irradiation. Because CE limits the level of hydrogen impurities in their fuel fabrication process, this methodology is found acceptable for application to the CE 16x16 fuel design up to a rod-average burnup of 60 MWd/kgM.

The major issue for external hydriding at extended burnup levels is an increase in hydriding that results in a decrease in cladding ductility reducing the threshold for cladding failure. The issue of decreased cladding ductility at the extended burnup level requested has already been discussed in Section 2.0(b) of this report and found to be acceptable for the CE 16x16 fuel design up to a rod-average burnup of 60 MWd/kgM.

(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 this axial gap (i.e., flattening). Because of the large local strains that would result from collapse, the cladding is assumed to fail. It is a CE design basis that cladding collapse is precluded during the fuel rod and burnable poison rod design lifetime. This design basis is the same as that in the SRP and has been approved by the NRC (Reference 3). We conclude that this design basis is also acceptable for the CE 16x16 fuel design up to a rod-average burnup of 60 MWd/kgM.

Evaluation - The longer in-reactor residence times associated with the burnup extension requested for ANO-2 fuel will increase the amount of creep of an unsupported fuel cladding. Extensive post-irradiation evaluations (Reference 2) by CE have not shown any evidence of cladding collapse or large local ovalities in their fuel designs. This is primarily the result of their use of prepressurized rods and stable (non-densifying) fuel in current generation designs.

In addition, CE has performed several post-irradiation examinations that have looked for axial gap formation in their modern fuel designs and concluded that the largest measured gaps are much smaller than those required to achieve cladding collapse for current CE fuel designs at a rod-average burnup of 60 MWd/kgM (Reference 1). These CE measured cold axial gaps have been corrected to hot axial gaps in the fuel rod during in-reactor operation for the cladding collapse analysis. The resulting hot gap used in the cladding collapse analysis is in excess of that expected at a 95% probability and a 95% confidence level based on a CE statistical analysis of the hot gaps (Reference 9). This cladding collapse analysis has demonstrated that the CE 16x16 cladding will not collapse at a rod-average burnup greater than 60 MWd/kgM. Therefore, ANO-2 has proposed that they no longer be required to address cladding collapse for new cores or reload batches of the CE 16x16 design unless design or manufacturing changes are introduced which would significantly reduce cladding collapse times for this fuel design. We conclude that this proposed approach is acceptable for future CE cores or reload batches of the 16x16 design with the requirement that the issue of cladding collapse be reevaluated should rod-average burnups exceed 60 MWd/kgM.

(c) Overheating of Cladding

Bases/Criteria - The design limit for the prevention of fuel failures due to overheating is that there will be at least a 95% probability at a 95% confidence level that the departure from nucleate boiling ratio (DNBR) will not occur on a fuel rod having the minimum DNBR during normal operation and AOOs. This design limit is consistent with the thermal margin criterion in Section 4.2 of the SRP, and thus, has been found acceptable for application to CE fuel designs (Reference 2). This design limit is not impacted by the proposed extension in burnup. Therefore, we conclude that this design limit remains acceptable for application to the CE 16x16 fuel design up to a rod-average burnup of 60 MWd/kgM.

Evaluation - As stated in Section 4.2 of the SRP, adequate cooling is assumed to exist when the thermal margin criterion to limit the DNBR or boiling transition in the core is satisfied. The analysis methods employed to meet the DNBR design basis are provided in References 30 through 34. These analysis methods have been approved by NRC for current burnup levels and are also found to be acceptable for application to the CE 16x16 design up to a rod-average burnup of 60 MWd/kgM.

The impact of rod bowing on DNB for the CE 16x16 design in ANO-2 has been addressed in Reference 35. We conclude that ANO-2 has adequately addressed the issue of cladding overheating for the CE 16x16 design up to a rod-average burnup of 60 MWd/kgM.

(d) Overheating of Fuel Pellets

Bases/Criteria - As a second method of avoiding cladding failure due to overheating, CE precludes centerline fuel pellet melting during normal operation and AOOs. This design limit is the same as given in the SRP and has been approved for use at current levels. We conclude that this design limit is also acceptable for the CE 16x16 fuel design up to a rod-average burnup of 60 MWd/kgM.

Evaluation - The design evaluation of the fuel centerline melt limit is performed with the approved CE fuel performance code, FATES3B (Reference 16). This code is also used to calculate initial conditions for transients and accidents. As noted earlier, the FATES3B code is acceptable for fuel performance calculations up to a rod-average burnup of 62 MWd/kgM (Reference 17).

In the CE centerline melting analysis, the melting temperature of the UO_2 is assumed to be 5080°F unirradiated and is decreased by 58°F per 10 MWd/kgM. This relation has been almost universally adopted by the industry and has been previously accepted by the NRC (Reference 3). Recent UO_2 fuel melting data with burnups to 30 MWd/kgM by Komatsu have shown no discernible decrease in melting temperature with burnup, and a drop of approximately 20°F per 10 MWd/kgM for UO_2-20% PuO with burnups up to 110 MWd/kgM (Reference 36). This demonstrates the conservatism employed by CE in their fuel melting temperature analysis at extended burnup levels. Therefore, we conclude that the ANO-2/CE analysis methods for fuel melting are acceptable for application to the CE 16x16 fuel design up to a rod-average burnup of 60 MWd/kgM.

(e) Excessive Fuel Enthalpy

Bases/Criteria - The SRP guidelines for a severe reactivity initiated accident (RIA) in a PWR, Section 4.2.II.A.2(f), state that for "all RIAs in a PWR, the thermal margin criteria (DNBR) are used in a fuel failure

criteria to meet the guidelines of Regulatory Guide 1.77 (Reference 37) as it relates to fuel failure." ANO-2/CE has adopted this criterion for fuel failure in addition to other more stringent criteria for RIAs (Reference 38). These criteria are still applicable to the burnup extension requested and therefore, are acceptable for application to the CE 16x16 fuel design up to a rod-average burnup of 60 MWd/kgM.

Evaluation - The NRC approved analysis methods for evaluating RIAs in CE plants is provided in Reference 39 and the specific analyses for ANO-2 are provided in Reference 38. The approved analysis methods described in Reference 39 are still applicable to the burnup extension requested and therefore, are acceptable for application to the CE 16x16 fuel design up to a rod-average burnup of 60 MWd/kgM.

The steady-state fuel operational data that are input to the CEA ejection analysis from the FATES3B code are dependent on fuel burnups. As noted earlier, the FATES3B code is acceptable for steady-state fuel performance applications for CE 16x16 fuel up to the 60 MWd/kgM rod-average burnup level requested in this submittal.

(f) Pellet/Cladding Interaction (PCI)

Bases/Criteria - As indicated in Section 4.2 of the SRP, there are no generally applicable criteria for 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 are used in CE fuel designs [see Sections 2.0(b) and 3.0(d)] and have been found to be acceptable in this application.

Evaluation - As noted earlier, CE uses the FATES3B code (Reference 16) to demonstrate that their fuel meets both the cladding strain and fuel melt criteria. This code has been found to be acceptable for these applications [see Sections 2.0(b) and 3.0(d)] and therefore, is acceptable for evaluating PCI failures for CE 16x16 fuel designs up to a rod-average burnup of 60 MWd/kgM.

CE has also presented PCI power ramping tests on fuel rods that are similar to their fuel designs up to rod-average burnups of approximately 48 MWd/kgM that demonstrate that the ramp terminal power level for fuel failure does not decrease with increased burnup. In addition, the maximum power capability of extended burnup fuel is reduced because of fissile material burnout, therefore, limiting the driving force for PCI failures. Consequently, we believe that CE 16x16 fuel designs have adequate PCI resistance up to a rod-average burnup of 60 MWd/kgM.

(g) Cladding Rupture

Bases/Criteria - Zircaloy cladding will burst (rupture) under certain combinations of temperature, heating rate, and differential pressure; conditions that occur during a LOCA. While there are no specific design criteria in the SRP associated with cladding rupture, the requirements of Appendix K to 10 CFR Part 50 must be met as those requirements relate to the incidence of rupture during a LOCA; therefore, a rupture temerature correlation must be used in the LOCA emergency core cooling system (ECCS) analysis. These Appendix K requirements for cladding rupture are not impacted by ANO-2's request to extend rod-average burnup to 60 MWd/kgM and therefore, we conclude that these requirements remain applicable to CE 16x16 fuel designs up to the burnup level requested.

Evaluation - An empirical cladding creep model is used by CE to predict the occurrence of cladding rupture in their LOCA-ECCS analysis. The rupture model is directly coupled to the cladding ballooning and flow blockage models used in the NRC approved ECCS evaluation model described in Reference 40.

The CE cladding rupture model is not affected by ANO-2's request to extend their burnup limit. Therefore, we conclude that the CE model for cladding rupture for LOCA-ECCS analyses is acceptable for application to the CE 16x16 fuel design up to a rod-average burnup of 60 MWd/kgM.

Another concern raised during previous high-burnup reviews (Reference 27), is that these higher burnups can result in fuel rod pressures that exceed system pressure and these higher fuel rod pressures can affect cladding rupture during a LOCA. For those CE fuel reloads that have calculated peak rod pressures above system pressure, CE has previously agreed (Reference 27) to reevaluate their LOCA-ECCS analyses to determine the most limiting LOCA conditions for these reloads. Therefore, we conclude that CE has addressed the issue of fuel rod pressures exceeding system pressure on cladding rupture in the LOCA-ECCS analysis.

Those important parameters that are input to the rupture analysis that can be burnup dependent, such as rod pressures, fission gas release, fuel stored energy, and gap conductance are calculated with the NRC approved code FATES3B. As noted earlier, the FATES3B code has been verified with data up to rod-average burnups of 60 MWd/kgM. Therefore, we conclude that the use of the FATES3B code is acceptable for input to LOCA-ECCS analyses of the CE 16x16 fuel design up to a rod-average burnup of 60 MWd/kgM, as requested in this submittal.

(h) Mechanical Fracturing

Bases/Criteria - Mechanical fracturing of a fuel rod could potentially arise from an externally applied force such as a hydraulic load or a load derived from core-plate motion. To preclude such failure, the applicant has stated (Reference 2) that fuel rod fracture stress limits shall be in accordance with the criteria given in Table 9-1 of CENPD-178 Revision 1 (Reference 41).

The review of CENPD-178, Revision 1, and the criteria given in Table 9-1 (Reference 41), has been completed and found acceptable by NRC for current burnup levels (Reference 3). The CE fracture stress limits in Reference 41 are conservatively based on unirradiated Zircaloy properties and are judged to remain conservative up to a rod-average burnup of 60 MWd/kgM for the mechanical fracturing analysis. Consequently, these criteria are also found to be acceptable for application to the CE 16x16 design up to a rod-average burnup of 60 MWd/kgM. However, future requests to extend the burnup beyond 60 MWd/kgM should be accompanied with measured cladding yield and fracture strength data to demonstrate that the rod fracture stress limits described in Reference 41 remain conservative up to the burnup level requested.

Evaluation - The mechanical fracturing analysis is done as a part of the seismic-LOCA loading analysis. A discussion of the seismic-LOCA loading analysis is given in Section 4.0(d) of this report.

4.0 FUEL COOLABILITY

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

(a) Fragmentation of Embrittled Cladding

Bases/Criteria - The most severe occurrence of cladding oxidation and possible fragmentation during an accident is a result of a significant degree of cladding oxidation during a LOCA. In order to reduce the effects of cladding oxidation for a LOCA, CE uses an acceptance criteria of 2200°F on peak cladding temperature and a 17% limit on maximum cladding oxidation as prescribed by 10 CFR 50.46. These criteria provided by CE for the LOCA analysis are acceptable for application to the CE 16x16 fuel design up to a rod-average burnup of 60 MWd/kgM.

Evaluation - The NRC approved cladding oxidation models in Reference 42 are used by CE to determine that the above criteria are met, as a result of the LOCA analysis. These models are not affected by the proposed extended burnup operation; however, the steady-state operational input provided to the LOCA analysis is burnup dependent. As noted earlier, those burnup dependent parameters important to the LOCA analysis, such as stored energy, gap conductance, fission gas release, and rod pressures from steady-state operation, are provided by the FATES3B code (Reference 16). Also, as noted earlier, FATES3B is acceptable for providing input to the evaluation of LOCA up to the requested rod-average burnup of 60 MWd/kgM.

The use of Reference 41 is also acceptable for evaluating cladding oxidation and fragmentation during a LOCA for the CE 16x16 fuel up to the rod-average burnup level requested in this submittal.

(b) Violent Expulsion of Fuel Material

Bases/Criteria - In a CEA ejection accident, 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 CEA ejection, Regulatory Guide 1.77 recommends that the radially-averaged energy deposition at the hottest axial location be restricted to less than 280 cal/g. This limit has been explicitly evaluated for ANO-2 in Reference 38 and the 280 cal/g limit remains acceptable up to a rod-average burnup of 60 MWd/kgM.

Evaluation - The CEA ejection analysis methods used by ANO-2/CE are described in the NRC approved report in Reference 39. The CEA ejection analysis for ANO-2 that utilizes the methods in Reference 39 are provided in Reference 38. In general, the most limiting assemblies in a CEA ejection accident are low burnup assemblies because these assemblies have the greatest power and, therefore, enthalpy capability in the core. The maximum enthalpies for fuel at a rod-average burnup of 60 MWd/kgM will be significantly bounded by the low burnup assemblies because power capability of this high burnup fuel is low. Consequently, fuel at extended burnup levels is expected to remain well below the 280 cal/g limit. We conclude that the analysis methods used by ANO-2/CE for evaluating the CEA ejection accident are acceptable for application to the CE 16x16 fuel up to the rod-average burnup requested in this submittal.

(c) Cladding Ballooning and Flow Blockage

Bases/Criteria - In the LOCA-ECCS analyses of CESSAR plants, empirical models are used to predict the degree of cladding circumferential strain and assembly flow blockage at the time of hot-rod and hot-assembly burst. These models are each expressed as functions of differential pressure across the cladding wall. There are no specific design limits associated with ballooning and blockage, and the ballooning and blockage models are integral portions of the ECCS evaluation model. We conclude that ANO-2 has addressed this issue in their LOCA-ECCS evaluation (Reference 40).

Evaluation - The cladding ballooning and flow blockage models used in the CE LOCA-ECCS analysis described in Reference 40 are directly coupled to the models for cladding rupture temperature and burst strain [discussed in Section 3.0(c)]. The CE cladding deformation, rupture, and flow blockage models used in Reference 40 are the same as those proposed by NRC in NUREG-0630 (Reference 43). These models are not affected by the burnup

extension requested in this submittal and therefore, Reference 40 remains acceptable for application to the CE 16x16 fuel design up to the rod-average burnup requested in this submittal.

The steady-state operational input that is provided to the LOCA analysis from the FATES3B fuel performance code (Reference 16) is burnup dependent. As noted earlier [see Section 3.0(g)], the FATES3B code has been verified against data to rod-average burnups of 62 MWd/kgM and previously approved for extended burnup application to the LOCA analysis (Reference 17). Therefore, this code is also acceptable for use in providing input to LOCA analyses of the CE 16x16 fuel design up to a rod-average burnup of 60 MWd/kgM.

(d) Structural Damage from External Forces

Bases/Criteria - To withstand the mechanical loads of a LOCA or an earthquake, the fuel assembly is designed to satisfy the stress criteria listed in Table 5-1 of Reference 41, and guide-tube deformation is limited such as to not prevent CEA insertion during the safe shutdown earthquake (SSE). These criteria have been found acceptable (Reference 3) for current burnup fuel and are also found acceptable for CE 16x16 fuel designs up to a rod-average burnup of 60 MWd/kgM.

Evaluation - The CE methods used to evaluate the mechanical loads due to a combined seismic-LOCA event are described in Reference 41. It is noted that the seismic-LOCA analyses are not affected by an increase in rod-average burnup up to 60 MWd/kgM and, therefore, previous bounding seismic-LOCA analyses remain applicable at this burnup level. This report has been approved by the NRC for current burnup levels and remains applicable for the CE 16x16 fuel design up to a rod-average burnup of 60 MWd/kgM.

5.0 DESIGN BASIS ACCIDENT ANALYSIS RELATIVE TO EXTEND FUEL BURNUP

The licensee has requested authorization to allow fuel burnup up to 60 MWd/kgM. The staff and licensee evaluated the potential impact of this change on the radiological assessment of design basis accidents (DBA) which were previously analyzed in the licensing of ANO-2.

The licensee, in discussions with the staff, concluded that the design basis accidents previously analyzed in their FSAR bound any potential radiological consequences of DBA that could result with the extended fuel burnup.

The staff reviewed a publication which was prepared for the NRC entitled, "Assessment of the Use of Extended Burnup Fuel in Light Water Reactors," NUREG/CR 5009, February 1988. The NRC contractor, the Pacific Northwest Laboratory (PNL) of Battelle Memorial Institute, examined the changes that could result in the NRC DBA assumptions, described in the various appropriate SRP sections and/or Regulatory Guides, that could result from the use of

extended burnup fuel (up to 60 MWd/kgM). The staff agrees that the only DBA that could be affected by the use of extended burnup fuel, even in a minor way, would be the potential thyroid doses that could result from a fuel handling accident. PNL estimates that I-131 fuel gap activity in the peak fuel rod with 60 MWd/kgM burnup could be as high as 12%. This value is approximately 20% higher than the value normally used by the staff in evaluating fuel handling accidents (Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facilities for Boiling and Pressurized Water Reactors").

PNL concluded in their report that for fuel damage accidents, "The percentage of fission-product inventory released from the fuel would not likely change as a result of the extended burnup; however, the fission-product inventory in the fuel would change for the long half-life fission products and actinides.... " PNL also concluded that the actinides would only minimally contribute to doses compared to the fission products and that the main concern for the actinides would be from the long-term effects of inhalation (lung dose) and ingestion of food products (vegetables, milk, and meat) raised in, or fed on food grown in contaminated soil. PNL concluded that the inventory of fission products, cesium-137 and strontium-90 would increase by a factor of almost 2 in the extended burnup fuel. However, the staff has concluded that their contribution to dose would be minimal.

For the fuel handling accident, PNL concluded that the use of Regulatory Guide 1.25 procedures for the calculation of accident doses for extended burnup fuel may be utilized. These procedures give conservative estimates for noble gas release fractions that are above calculated values for peak rod burnups of 60 MWd/kgM. Iodine-131 inventory, however, may be up to 20% higher than that predicted by Regulatory Guide 1.25 procedures.

The staff, therefore, reevaluated the fuel handling accidents for the ANO-2 facility with an increase in iodine gap activity in the fuel damaged in a fuel handling accident. Table 1 presents the fuel handling accident thyroid doses presented in the operating licensing Safety Evaluation Report, dated November 1977, and in the Supplemental SERs dated March and September 1978, and the increased thyroid doses (by 20%) resulting from extended burnup fuel.

Table 1

Thyroid Doses as a Consequence of DBA Fuel Handling Accidents

	Exclusion Two Hour Boundary		Low Population Zone	
	Thyroid Dose (Rem)		Thyroid Dose (Rem)	
Fuel Handling Accident	A*	B**	A*	B**
Spent Fuel Area	35	42	3	3.6
Containment Building ***	<35	<42	<3	<3.6

*A SER/SSER #2 dose
**B Extended fuel burnup dose
***SER Supplement 1 dated March 1978 indicated that consequences of this accident are bound by the consequences of a fuel handling accident in the spent fuel area.

The staff concludes that the only potential increased doses that could result from DBA with extended fuel burnup to 60 MWd/kgM is the thyroid dose resulting from fuel handling accidents and these doses remain well within the 300 Rem thyroid exposure guideline values set forth in 10 CFR Part 100 and that this small calculated increase is not significant.

C. EMERGENCY CIRCUMSTANCES

[As stated in the licensee's application for amendment.] the requested changes constitutes an emergency situation pursuant to 10CFR50.91(a)(5) because: (1) absent NRC action by November 28, 1990, ANO-2 must be shutdown; (2) this emergency situation could not have been avoided by Entergy Operations; and (3) the proposed change does not involve a significant hazards consideration.

1. Current Condition

ANO-2 currently is operating at 100 percent power in Cycle 8 and has accumulated approximately 332 effective full power days (EFPDs) as of November 21, 1990. Entergy Operations has calculated that at approximately 340 EFPDs, currently estimated to occur on November 29, 1990, continued operation of ANO-2 will be precluded because the facility will reach the rod average fuel burnup limit of 52 MWD/Kg. The current condition of the facility cannot be rectified absent the proposed change to the license or plant shutdown and refueling (not presently scheduled until February 1991).

2. Time Constraints

The NRC first requested that a license amendment be submitted by Entergy Operations on November 15, 1990. Because the change proposed in this submittal must be reviewed and approved by the NRC prior to November 29, 1990, the 30 day notice and comment provisions of 10CFR50.91(a)(2) cannot be met. Accordingly, Entergy Operations has developed this request for issuance of a license amendment pursuant to the emergency provisions of 10CFR50.91(a)(5). This request has been submitted in a timely manner considering the need to develop a significant hazards evaluation and the need to support the emergency request.

Moreover, Entergy Operations initially had requested NRC review and approval of the methodology to evaluate an increase to the ANO-2 fuel pin burnup limit on July 20, 1989. Since that submittal, Entergy Operations maintained active communication with the NRC to monitor the staff review of the request (see Entergy Operations letters to the staff in May and September, 1990 to address specific NRC technical questions). Not until the NRC staff requested a license amendment on November 15, 1990, was there an indication of the need for such an amendment. Therefore, Entergy Operations has acted in a timely fashion with this submittal which provides the NRC staff with adequate time to process an emergency change in accordance with 10CFR50.91(a)(5).

3. Hardship Absent Relief

Without NRC approval of this emergency request, ANO-2 must shutdown and either await completion of the standard license amendment process, or change the fuel in the facility to permit continued operation. These options present hardship to Entergy Operations which are outweighed by the approval of the emergency request, especially considering the absence of a significant hazards associated with the proposed change. ...

4. ...

5. Plan for Compliance

ANO-2 is currently in compliance with the applicable requirements of the operating license and Technical Specifications and will continue to maintain compliance with these and any other requirements. With the approval of the proposed change, continued operation of ANO-2 beyond 340 EFPDs will be possible and specifically permitted; hence, at no time does Entergy Operations anticipate non-compliance.

...

Based on the above, the staff has determined, pursuant to 10 CFR 50.91(a)(5), that failure to act in a timely manner will result in plant shutdown. Further, the licensee maintained communication with the NRC staff and promptly submitted its amendment request when it was determined such action was warranted. Accordingly, the Commission has det rmined that emergency circumstances exist which warrant prompt action by the Commission.

D. FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards considerations, if operation of the facility, in accordance with the amendment would not:

(1) Involve a significant increase in the probability or consequences of any accident previously evaluated; or

(2) Create the possibility of new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

This amendment has been evaluated against the standards in 10 CFR 50.92 and does not involve any significant hazards considerations. The following excerpt from the licensee's submittal lists these criteria and the licensee's description:

Criterion 1 - Does not Include a Significant Increase in the Probability or Consequence of an Accident Previously Evaluated.

The effects of extended burnup up to 60 MWd/kg have been evaluated in the the [sic] Combustion Engineering ... [CE] Report CEN-386-P with respect to the previously identified 21 fuel performance topics that were judged to be burnup dependent and/or important in determining the behavior of extended burnup fuel. Using the results of this [CE] Report, it was concluded that the fuel performance characteristics do not significantly change with extended burnup up to 60 MWd/kg and with the exception of the fuel handling accident, no change in consequences of a design basis accident is expected.

With respect to the fuel handling accident, extended burnup will result in fewer fuel movements over the life of the plant in comparison to lower burnup fuel management schemes and thus a decrease in the probability of an accident occurrence. The consequences of a fuel handling accident are also not significantly affected. The effect of extended burnup with respect to offsite dose consequences as a result of a fuel handling accident has been previously evaluated by the NRC in NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors." This report concludes that there would be a slight increase (by 20%) in

thyroid doses resulting from increased Iodine 131 gap activity from burnups to 60 MWd/kg. The resulting doses are small fractions of the applicable regulatory requirements of 10CFR Part 100 as concluded in Calvert Cliffs Safety Evaluation Report of January 10, 1990.

Criterion 2 - Does not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated

Since the early 1980's, significant data have been accumulated on the effects of high burnup on fuel. This data and analytical techniques have been utilized to project the effects of high burnup in support of this amendment. The measured and projected effects show the fuel will continue to exhibit stable predictable performance. Therefore, no new or different kind of accident will be created.

Criterion 3 - Does not Involve a Significant Reduction in the Margin of Safety

The [CE] Report in support of this amendment has evaluated the 21 fuel performance topics that were judged to be burnup dependent and/or important in determining the behavior of extended burnup fuel. This evaluation for each cycle concluded adequate margins of safety continue to be provided with fuel burnup to 60 MWd/kg.

Accordingly, the Commission has determined that this amendment involves no significant hazards considerations.

E. STATE CONSULATION

In accordance with the Commission's regulations, efforts were made to contact the Arkansas State representative. The state representative was contacted and had no comments.

F. ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an Environmental Assessment and Finding of No Significant Impact has been prepared and published in the Federal Register on November 14, 1990 (55 FR 47593). Accordingly, based upon the environmental assessment, the Commission has determined that the approval of the extended fuel burnup limit for ANO-2 will not have a significant effect on the quality of the human environment.

G. CONCLUSIONS

We have reviewed the ANO-2 request, as submitted in Reference 1, to extend the burnup level of the CE 16x16 fuel design to a rod-average burnup of 60 MWd/kgM in accordance with the SRP, Section 4.2. We conclude that this request by ANO-2, is acceptable. However, it should be stressed that future requests to

extend the rod-average burnup limit beyond 60 MWd/kgM should be accompanied with corrosion, cladding strain, and yield and fracture strength data at the extended burnup levels requested. These data are necessary to support the irradiation of higher burnup fuel beyond 60 MWd/kgM.

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: November 27, 1990

Principal Contributors: S.L. Wu
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Topical Report
"Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/kgU for Combustion Engineering 16x16 PWR Fuel"

ABSTRACT

Several Utilities using Combustion Engineering 16x16 fuel assembly designs have implemented programs to extend their fuel cycle lengths from 12 to 18 months and beyond. The maximum 1-pin burnup predicted for these extended burnup cycles exceeds the 52 MWD/kg limit presented in the existing C-E Extended Burnup Operation topical report. For example, Arkansas Nuclear One Unit 2 Cycle 8 (the third 18-month cycle for that unit) will have a number of fuel pins that exceed this current burnup limit. This report verifies the adequate modelling of those 16x16 fuel design pins to 60 MWD/kg (the new limit required by the implementation of longer fuel cycles) by supplementing the existing topical report with additional data and discussions. The conclusions of this report regarding fuel assembly length change and shoulder gap change are applicable to Combustion Engineering 16x16 fuel assembly designs employing [ ].

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INTRODUCTION

Several Utilities using Combustion Engineering 16x16 fuel assembly designs have implemented programs to extend their fuel cycle lengths from 12 to 18 months and beyond. The predicted maximum 1-pin burnup for these extended burnup cycles exceeds the current C-E 1-pin burnup limit presented in Reference 1, 52 MWD/kg. For example, Arkansas Nuclear One Unit 2 Cycle 8 (the third 18-month cycle for that Unit) will have a number of fuel pins that exceed this current burnup limit. This report justifies a 1-pin burnup limit of 60 MWD/kg for 16x16 fuel assembly designs by supplementing Reference 1 with data and discussions covering the additional burnup range required by the implementation of longer cycles, 52 MWD/kg to 60 MWD/kg.

Reference 1 also specified a limit on batch average discharge burnup. However, a review of the various burnup dependent fuel performance topics discussed in Reference 1 indicated no explicit dependence on batch average burnup. Therefore, the C-E batch average discharge burnup limit of Reference 1 has been deleted.

Reference 1 presented data and/or discussions on 21 fuel performance topics that were judged to be burnup dependent and/or important in determining the behavior of fuel at extended burnup. The existing data and discussions presented in Reference 1 support the acceptability of a 1-pin burnup limit of 60 MWD/kg for the following 8 fuel performance topics: fatigue of the fuel rod, fuel rod bowing, fuel rod fretting wear, cladding deformation and rupture, guide tube wear, fuel assembly holddown, grid irradiation growth and spacer grid relaxation. Consequently, only a short discussion is provided for each of these topics.

The remaining 13 fuel performance topics are discussed within update sections that present the additional data and/or discussions needed to support the acceptability of a 1-pin burnup limit of 60 MWD/kg.

The conclusions of this report regarding fuel assembly length change and shoulder gap change are applicable to Combustion Engineering 16x16 fuel assembly designs employing [

].

DISCUSSION

The contents of the following update sections generally follow the format of their respective section in Chapter 3 or 4 of Reference 1. Each (sub)section is numbered identically to its respective (sub)section in Reference 1 with the addition of ".a". Each section has an introduction which specifies how the succeeding subsections should be treated, i.e., whether they append or replace their respective subsection. The figures, tables and references of each section are numbered sequentially in the following form, "section #"-"sequence #", e.g., 4.1.3.a-1, with the exception of Reference 1 which is a general reference that applies to all sections of this report.

3.3.6.a <u>Cladding Collapse</u>

This section replaces Section 3.3.6 of Reference 1.

Collapse is the term applied to a condition where a slightly oval cladding tube will "flatten" into a significant axial gap in its fuel or poison pellet column. The conditions leading to collapse are long term phenomena since collapse occurs only after the cladding has crept into an oval shape from its nearly circular shape at beginning of life. The driving force for this creep is supplied by the differential pressure across the fuel or poison rod cladding.

C-E design characteristics which mitigate cladding collapse are:

- o Fuel and poison rods are prepressurized with helium which offsets the effects of external pressure to the extent that cladding long term creep and cladding ovalization are reduced.

- o "Non-densifying" or stable fuel pellets are used to prevent the formation of significant axial gaps within the fuel column. This allows the fuel pellets to support the cladding later in life when the fuel-cladding gap closes.

- o Poison rods behave in a similar fashion to fuel rods except the pellets are not subject to densification.

The cladding collapse model is discussed in Section 4.1.4.a.

4.1.1.a Fatigue

The discussion provided in Section 4.1.1 of Reference 1 applies to the proposed increase in the 1-pin burnup limit to 60 MWD/kg. The method used to calculate fatigue damage is applicable to extended burnup operation since the other sections of this report show that the individual components of the method (e.g., cladding creep and fuel swelling) are adequately modeled and the cladding has adequate ductility.

4.1.2.a Cladding Corrosion

The following subsections append the corresponding subsections of Reference 1.

4.1.2.1.a Corrosion Behavior

Oxide thickness data from three C-E PWRs, Calvert Cliffs-1, Ft. Calhoun, and ANO-2, for rod average burnups of up to [ ] MWD/kg have recently become available (References 4.1.2.a-1, 4.1.2.a-2, 4.1.2.a-3, 4.1.2.a-4). The maximum burnup rod for which oxide thickness data are available for the 14x14 design is approximately [ ] MWD/kg (Reference 4.1.2.a-4) and for the 16x16 design (Reference 4.1.2.a-3) is approximately 58 MWD/kg. The recent high-burnup oxide thickness data are presented together with the data of Figure 4-3 of Reference 1 in Figure 4.1.2.a-1. The U_{235} enrichment level for these high-burnup rods was between 3.03 and 4.00%. The U_{235} enrichment for future fuel batches is expected to increase, but the burnups are not expected to exceed 60 MWD/kg. The available oxide thickness data on irradiated fuel cladding approximately covers the maximum burnup level of future high burnup rods. As a first approximation, the oxide thickness at a burnup of 60 MWD/kg was estimated from a regression fit to the 16x16 (ANO-2) oxide thickness data. Regression analysis of the 16x16 (ANO-2) oxide data resulted in a best estimate oxide thickness of [ ] microns at 60 MWD/kg and an upper bound ($\bar{x} + 3\sigma$) oxide thickness of [ ] microns at 60 MWD/kg. A similar fit to the Calvert Cliffs-1 14x14 data yields a best-estimate thickness of [ ] microns and an upper bound of [ ] microns.

Recently published high-burnup corrosion data from other PWRs (References 4.1.2.a-7 to 4.1.2.a-10) are presented together with the data from Figure 4-4 of Reference 1 in Figure 4.1.2.a-2. It is worthwhile to note that the corrosion data presented in Figure 4.1.2.a-2 refers to fuel rods with fuel enrichments lower than 4% U_{235} and several irradiation cycles of the order of 12 months duration. The heat rates of these fuel rods are generally lower than those expected for future high-burnup fuel rods.

Nevertheless, for a rod average burnup of ~ 60 MWD/kg, the upper limit of expected oxide thickness from Figure 4.1.2.a-2 is about 100 microns. This is in reasonable agreement with the upper bound estimate presented above for the ANO-2 data [ ] and the Calvert Cliffs-1 data [ ].

Another important aspect of cladding corrosion is the extent of hydrogen uptake by the cladding. A fraction of the amount of hydrogen liberated by the Zircaloy corrosion reaction is absorbed by the cladding. As discussed in Section 4.1.5.a, the absorbed hydrogen may reduce the ductility of the cladding. Hydrogen concentrations measured on cladding specimens from several PWR fuel rods are presented in Figure 4.1.2.a-3. A detailed analysis of the data (Reference 4.1.2.a-11) shows that a pickup fraction of 18% represents a reasonable upper limit on hydrogen absorption by cladding at high burnups. This pickup fraction translates to a cladding hydrogen level of about [ ]. The relationship between hydrogen level and cladding ductility is further discussed in Section 4.1.5.2.a.

4.1.2.2.a <u>Evaluation of Cladding Corrosion at Extended Burnup</u>

Based on the limitations discussed in Section 4.1.2.1.a, the 3σ upper bound oxide thickness is estimated to be about [ ] microns for fuel cladding at a rod average burnup of 60 MWD/kg. The cladding wastage due to this level of oxide layer thickness is insignificant with regard to cladding stresses. Although some tendency towards [oxide spalling] was reported at a level of oxide layer thickness of about [ ] microns (References 4.1.2.a-4 and 4.1.2.a-7), fuel rod integrity was not impaired. It is, therefore, concluded that cladding corrosion is not likely to impair the integrity of fuel rods irradiated to rod average burnups of 60 MWD/kg.

An oxide layer will, of course, increase the surface temperature of the cladding. For example, the maximum local temperature increase at the metal-oxide interface due to a 3σ upper bound oxide layer (on the 16x16 or 14x14 design fuel rods), assuming a local fuel rod linear heat rate of [ ] kw/ft, is calculated to be about [ ]. On a rod-average

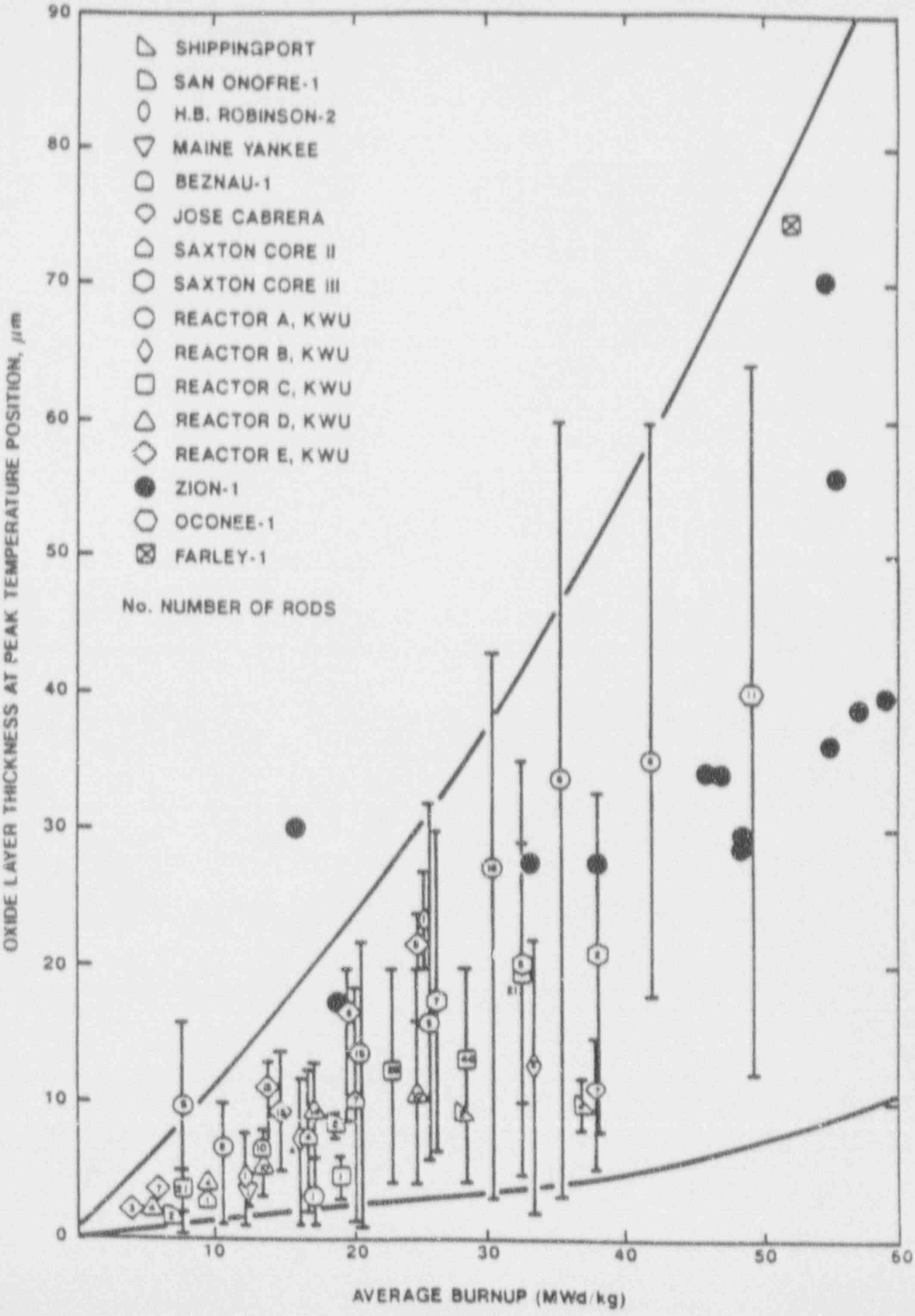
basis, the temperature increase at the metal-oxide interface will be considerably less. The largest impact of the insulating oxide layer occurs at end-of-life when the linear heat rate of the fuel rod is significantly lower than the linear heat rate of the peak rod in the core. Thus, it is concluded that the effect of oxide build-up on fuel temperature and stored energy is essentially counteracted by the lower linear heat rates that occur towards end-of-life. Thus, corrosion, based on observed oxide-thicknesses at 60 MWD/kg in operating reactors, will not be limiting.

However, additional factors in the future must be considered. Specifically, the factors that need to be considered are EFPD (corresponding average-linear heat rate) to achieve 60 MWD/kg and the reactor coolant conditions (temperature and chemistry). If these factors differ substantially from the data base, additional corrosion evaluations would be warranted. These factors will be monitored and corrosion evaluations performed as necessary, particularly if 1) the EFPD to achieve maximum burnup are considerably shorter, 2) the reactor inlet temperatures are considerably higher, or 3) the coolant lithium level is significantly higher than the ranges covered in the current data base from the operating reactors. In addition, the impact of cladding changes that optimize composition and processing history to improve the in-reactor corrosion resistance compared to that used in the current data base will also be included in the above evaluations.

ROD AVERAGE BURNUP, MWD/kg

MAXIMUM OXIDE THICKNESS, microns

Figure 4.1.2.a-2
Cladding Peak Oxide Thickness as a Function of Average Burnup



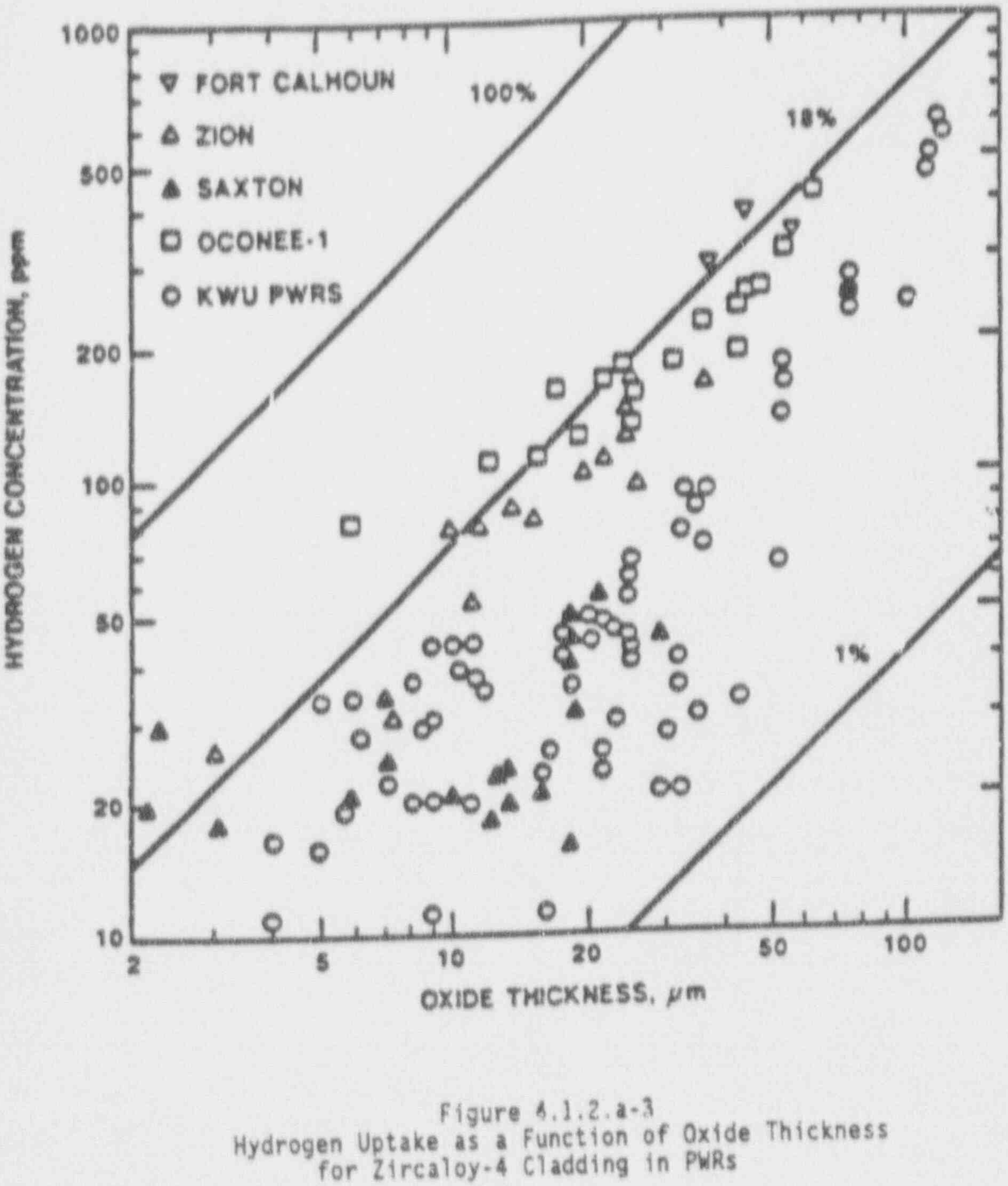


Figure 4.1.2.a-3
Hydrogen Uptake as a Function of Oxide Thickness
for Zircaloy-4 Cladding in PWRs

4.1.3.a Cladding Creep

Append Subsection 4.1.3.3 of Reference 1 with the following material:

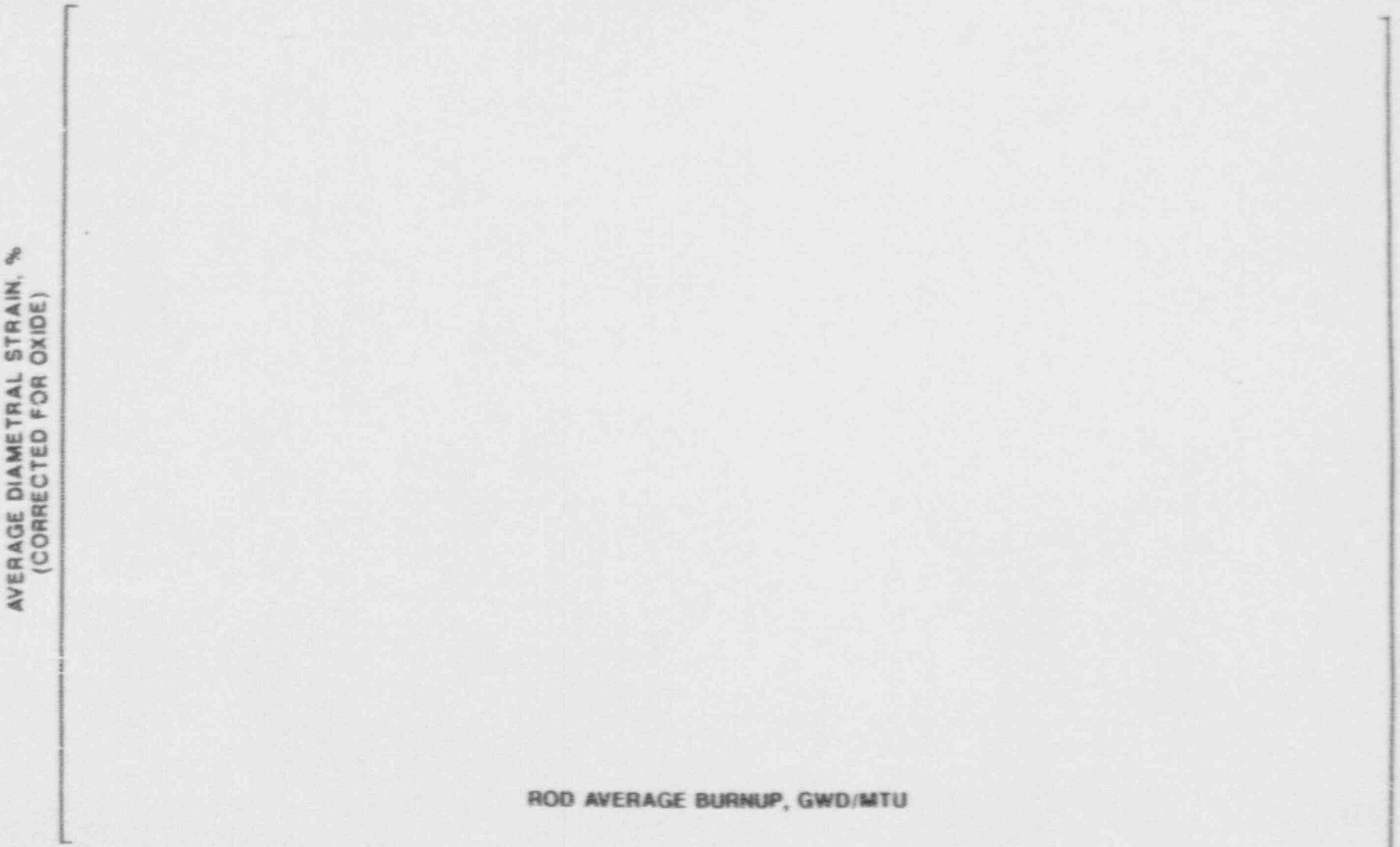
4.1.3.3.a Evaluation of Creep

Diametral creep measurements are now available for several high burnup fuel rods irradiated in Calvert Cliffs-1 and Fort Calhoun (References 4.1.3.a-1 and 4.1.3.a-2). These data, corrected for the presence of oxide and converted to resulting diametral strain, are presented in Figure 4.1.3.a-1. The rod average burnups of these rods are [ ] Due to the contact between the fuel pellet and cladding at these high burnup levels, the fuel rod diametral strain is strongly influenced by the fuel pellet's swelling behavior. The data presented in Figure 4.1.3.a-1 show that the diametral behavior of the fuel rod is a continuous function to rod average burnups of 60 MWD/kg and that the model discussed in Reference 1 is adequate for 1-pin burnups of up to 60 MWD/kg.

The diametral strain data presented in Figure 4.1.3.a-1 show that the fuel rod diameter does not change significantly during extended burnup operation. Early-in-life, prior to the establishment of fuel-cladding contact, cladding creepdown occurred due to coolant pressure. [

]

Figure 4.1.3.a-1
Diameteral Strain of High Burnup Rods Irradiated
in Fort Calhoun and Calvert Cliffs-1



4.1.4.a Cladding Collapse

This section replaces Section 4.1.4 of Reference 1.

Cladding tubes generally have a minor degree of variation from a perfectly circular cross section with uniform wall thickness. When subjected to a net external pressure in the reactor, bending stresses are produced as a result of the slightly imperfect geometry. Under the high temperature and neutron flux conditions in the reactor, the Zircaloy cladding creeps in response to the bending stresses. The resulting creep strain increases the deviation from the circular shape, thereby increasing the bending stresses. This process continues at an increasing rate until contact is made with the pellets, or if a significant axial gap exists in the pellet column, until an unstable condition is reached and the cladding "collapses" into a distorted shape.

Observations indicate that no significant axial gaps form in the fuel pellet column during the operation of Combustion Engineering's modern design fuel, which has prepressurized fuel rods and stable "nondensifying" fuel pellets. Such gaps would be evidenced by unusual local ovalities of the fuel rod cladding, a distinct region of atypical crud deposition around the cladding circumference, or atypical signals during gamma scanning. None of these indications have been observed during the extensive post-irradiation examination programs conducted on both the 14x14 and 16x16 fuel designs.

It can be inferred from these post irradiation examinations of modern design C-E fuel that during hot full power operation the axial gaps in a fuel column are usually only a fraction of the length of a pellet. The gaps are measured in the cold condition. The largest cold gap measured in modern C-E fuel was 0.9 inches. It was calculated that thermal expansion of the fuel column during reactor startup reduces this cold gap to $\leq$ 0.3 inches. Thus, the largest hot gap inferred from all post irradiation examinations of modern C-E fuel was 0.3 inches. This conclusion is supported by the corrosion patterns observed during visual examinations.

4.1.4.1.a Modeling of Cladding Collapse

The current methods of evaluating resistance to cladding collapse are described in Reference 4-17 of Reference 1, and Reference 4.1.4.a-1. Reference 4-17 of Reference 1 describes a method which utilizes the CEPAN computer code to predict creep deformation and collapse time of Zircaloy cladding containing an initial ovality. Although large hot gaps have not been inferred for modern design C-E fuel, this method assumes that a gap in the pellet column exists at the most unfavorable elevation in the fuel rod. No credit is taken for the support offered by the pellets at the edges of the gap. This original method of selecting input to CEPAN resulted in a deterministic combination of the worst case cladding as-built dimensions and worst case operating conditions during the fuel lifetime. The NRC concluded that CEPAN provides an acceptable analytical procedure for determining the minimum time to collapse for C-E Zircaloy clad fuel. If this minimum collapse time exceeds the fuel lifetime, then collapse resistance has been demonstrated.

A modification of the above method is described in Ref. 4.1.4.a-1. This modification is applied to the normal CEPAN results to account for the support provided to the cladding by the pellets at the edges of the gap. The adjustment varies as a function of the length of the gap or unsupported cladding. As the gap considered becomes longer, the results approach the normal CEPAN results.

4.1.4.2.a Effect of Extended Burnup

Since cladding collapse is a creep-related phenomenon, the longer residence times associated with extended-burnup fuel will increase the amount of creep of unsupported cladding. The increased creep strain will be accounted for in the analysis of the ability of the fuel rod to resist cladding collapse.

4.1.4.3.a Evaluation of Cladding Collapse

Although early experience with densifying UO_2 fuel pellets indicated that cladding collapse could result in fuel failure, improvements in fuel design, notably the development of stable fuel pellet types and rod pressurization, have essentially eliminated this concern. Current commercial fuel pellets have shown through operating performance that significant axial gaps do not form in the fuel pellet column during operation. Without the occurrence of gaps of sufficient length, cladding collapse cannot occur and, as a consequence, the cladding will remain stable and will not be subject to high local strains from this effect. Furthermore, there is no evidence to indicate that continued operation of fuel rods having cladding in oval contact with the fuel pellet column is detrimental.

C-E has performed cladding collapse calculations with the modified method described in Section 4.1.4.1.a using very conservative input assumptions. The assumed length of the axial gap in the fuel column bounded the largest hot axial gap in modern C-E fuel (See Section 4.1.4.a). These calculations have shown that the predicted collapse times far exceed the longest residence time ever expected for C-E fuel that is operated to a maximum pin burnup of 60 MWD/kg. It has therefore been concluded that unless significant changes in design or manufacturing methods are introduced, modern C-E fuel and poison rods for both 16x16 and 14x14 designs are not susceptible to cladding collapse. On this basis, C-E will no longer specifically address cladding collapse for new cores or reload batches unless design or manufacturing changes are introduced which would significantly reduce predicted collapse time results. In the event such changes do occur, the modified method described in Section 4.1.4.1.a will be used to confirm that cladding collapse will not occur during the design lifetime of the fuel.

4.1.5.a Ductility of Fuel Cladding

This section replaces Section 4.1.5 of Reference 1.

Exposure of the fuel rod Zircaloy cladding to fast neutron irradiation causes the cladding material to strengthen and lose some of its ductility. In addition, the fuel rod Zircaloy cladding reacts with water during reactor operation to form a zirconium dioxide (ZrO_2) layer on the outer surface of the fuel rod. Hydrogen is produced by this reaction and a fraction of the liberated hydrogen (approximately 0.18) is absorbed by the cladding. This hydrogen uptake may also reduce the ductility of the cladding. The fuel rod design criteria related to strength and ductility were discussed in Sections 3.3.2 and 3.3.3 of Reference 1, respectively. Since the fuel rod design calculations are based on the yield strength of unirradiated cladding, the increase in the yield strength of cladding due to neutron irradiation does not pose a strength limitation on the cladding's performance. The loss of ductility due to the neutron irradiation and hydrogen uptake, however, needs to be evaluated to assure that adequate cladding ductility exists at extended burnup levels to ensure that the design strain limits remain valid. The effect of extended burnup operation on the cladding ductility is evaluated in this section.

The elevated temperature cladding strain design limit used in the C-E FSARs is 1%. A review of the mechanical property data of high fluence cladding (from fuel rods with rod average burnups up to 60 MWd/kg) [ ]. Since the deformation capability of irradiated cladding during the normal reactor operation and anticipated transients is important, the mechanical properties of irradiated Zircaloy-4 at the deformation temperatures of about 600°F were considered in the analysis of the extended burnup data. The combined effect of the neutron fluence and hydrogen uptake on the mechanical properties of Zircaloy-4 is evaluated below.

4.1.5.1.a Mechanical Properties of Irradiated Zircaloy at Extended Burnups

C-E has obtained data on the mechanical properties of Zircaloy-4 cladding irradiated in the Fort Calhoun reactor to local burnups of up to 62 MWd/kg (Reference 4.1.5.a-1). In addition, mechanical property data have also become available for fuel cladding irradiated in Oconee-1 (Reference 4.1.5.a-2) and Zion (References 4.1.5.a-3, 4.1.5.a-4) to extended burnups. These data were recently analyzed to evaluate the effects of irradiation and hydriding on the mechanical properties of Zircaloy-4 at high fluences (Reference 4.1.5.a-5). These data are described below together with the low burnup data presented in Section 4.1.5 of Reference 1.

C-E uses [ ] fuel rod cladding (Reference 4.1.5.a-6). The increase in elevated-temperature yield strength due to irradiation is illustrated in Figure 4.1.5.a-1 (References 4.1.5.a-7 through 4.1.5.a-10). An increase in yield strength has also been observed by CE at extended burnup (Reference 4.1.5.a-5). The increase in the ultimate tensile strength of irradiated Zircaloy due to higher hydrogen levels, on the other hand, does not appear to be significant (see Figure 4.1.5.a-2). The data of Evans and Parry (Reference 4.1.5.a-11) shown in this figure indicate that there is no change in the ultimate strength of irradiated Zircaloy-2 at temperatures above 100°C (210°F) when the hydrogen level is increased from 0 to 200 ppm. The yield strength behaves in a similar manner.

[

]

The fluence dependence of the [

] is illustrated in

Figure 4.1.5.a-3. The data (Reference 4.1.5.a-12) suggest that for [

] These tests were conducted at high strain rates.

It has been theoretically predicted by Nichols (Reference 4.1.5.a-13) and Ibrahim and Coleman (Reference 4.1.5.a-14) and experimentally verified by Ibrahim (Reference 4.1.5.a-15) and Wood (Reference 4.1.5.a-16) that at the lower strain rates more appropriate to the creep deformation rates of the fuel cladding, the uniform elongation is greater than estimated from the short term, high strain rate mechanical tests. Irradiation data for a low fluence (1.8 x 10^{21} n/cm^2) nickel free Zircaloy 2 (Reference 4.1.5.a-15) indicate that at a stress of 332 MPa, the creep rupture strain is greater than 5.1%. Two factors need to be considered for in-reactor creep of cladding with higher fluence. Firstly, at the lower stresses appropriate to in-reactor cladding creep, the creep strains at rupture are expected to be higher (Reference 4.1.5.a-13). Secondly, with an increasing level of fluence, the creep strain will decrease. Based on the available ductility data on high fluence cladding irradiated in power reactors, it is concluded that the cladding ductility at high burnups will be significantly greater than 1% as a result of the net effect of these two opposing factors.

4.1.5.2.a <u>Influence of Hydrogen on Mechanical Properties</u>

A fraction of the amount of hydrogen liberated by the Zircaloy corrosion reaction with the primary coolant is absorbed by the cladding. It remains in solution in the Zircaloy until the terminal solid solubility of hydrogen is exceeded. At 300°C (572°F), the solubility limit is approximately 100 ppm. Amounts in excess of the solubility limit will precipitate as zirconium hydride platelets.

It has been established that the ductility reduction due to hydrogen is dependent not only on the quantity of hydrides but also on their orientation. For example, if the hydrides are precipitated so that their major axis is parallel to an applied stress, the reduction in ductility is relatively small. [

]

Evans and Parry (Reference 4.1.5.a.11) determined the temperature above which the effects of unfavorably oriented hydrides disappear in cold-worked and stress-relief-annealed Zircaloy-2 cladding irradiated to low fluences. At temperatures above 200°C (392°F), adversely oriented hydrides up to 200 ppm did not influence the ductility as measured by the reduction in area (Figure 4.1.5.a-4). Watkins et al. (Reference 4.1.5.a-17) have conducted tests on cold-worked tubular samples of Zircaloy-2 prehydrided to levels of up to 800 ppm which have circumferentially oriented hydrides. Tensile tests showed that hydrogen concentration had only a minor effect on ductility at 300°C (572°F) (Figure 4.1.5.a-5). Specimens charged with hydrogen showed values of the reduction in area at failure in excess of 60%. Thus, it has been concluded that at elevated temperatures, circumferentially oriented hydrides up to 800 ppm do not influence the ductility of Zircaloy cladding irradiated to fluence levels of 10^{20} n/cm^2.

4.1.5.3.a <u>Combined Effect of Radiation Damage and Hydriding on the Ductility of Cladding at Extended Burnups</u>

The ductility of extended burnup fuel rod cladding with rod average burnups approaching 60 MWD/kg (local burnups to 62.5 MWd/kg corresponding to cladding fluence levels to 16.2 x 10^{21}, n/cm^2, E > 0.821 MeV) has been recently measured by axial and ring tension tests and diametral burst tests (References 4.1.5.a-1 to 4.1.5.a-4). The results of these mechanical tests

demonstrated the combined effects of neutron damage and hydrogen uptake on the mechanical properties of highly irradiated Zircaloy-4. The strain rates resulting from the load application in these tests were also significantly higher than the fuel rod cladding strain rates expected during normal steady-state operation and also during the anticipated operational transients of a power reactor. Ring tensile tests at 650°F on cladding from 5-cycle rods (rod average burnups 49.5 to 49.9 MWd/kg) irradiated in Oconee-1 (Reference 4.1.5.a-2) show uniform strains in the range of 2 to 3% and total strains in the range of 3.8 to 8.4%. Axial tension tests at 650°F on cladding from the same rods resulted in uniform strains in the range of 0.93 to 1.43% (average 1.29%) and total strains in the range of 5.68 to 15.31%. Therefore, the Oconee-1 cladding data indicate that at a burnup of about 50 MWd/kg, the cladding can withstand an additional strain of 1% prior to plastic instability and about 4% strain prior to failure.

Axial tension tests on six-cycle Fort Calhoun cladding (local burnups in the range of 57.6 to 63.3 MWD/kg) (Reference 4.1.5.a-1) show that for a deformation temperature range of 392 to 752°F, the uniform strains are in the range of 0.7 to 0.8% and total strains are in the range of 5 to 9%. Thus, Fort Calhoun cladding tensile data indicate that at an end-of-life burnup of approximately 60 MWD/kg, the cladding can withstand approximately 1% additional strain prior to the onset of plastic instability and at least 5% additional strain prior to failure.

Burst test data on high burnup cladding are available from fuel rods irradiated in Fort Calhoun and Zion. The burst test data on Zion cladding with a rod average burnup of 55.3 MWD/kg show total circumferential strains of 0.79 to 2.69% (Reference 4.1.5.a-3). At lower burnup levels of 38 and 46 MWD/kg, the Zion cladding burst test results show total circumferential strain values above 3% (Reference 4.1.5.a-4). Burst test data are available on Fort Calhoun cladding with rod average burnups approaching 60 MWD/kg (Reference 4.1.5.a-1). At a local burnup of 41.6 MWD/kg, the uniform strain values are 1.12 and 1.21% and total strain values are 6.9

and 5.6%. At a local burnup level of 52.3-53.2 MWD/kg, the uniform strains are 1.43 to 1.75% and total strains are 4.5 to 4.7%. At a local burnup level of 54.7 to 62.5 MWD/kg, the uniform strains are 0.03 to 0.11% and total strains are 1.24 to 4.19%.

The material ductility at 572 to 599°F as a function of fluence is shown in Figure 4.1.5.a-6 (Reference 4.1.5.a-5). For fluence values up to ~$9x10^{21}$ n/cm^2 (E > 0.821 MeV or $8x10^{21}$ n/cm^2 E > 1 MeV) (corresponding to burnups up to ~53 MWd/kg), [

] Moreover, based on a detailed analysis of the microstructures of the fractured specimens, the fracture mode at burnups greater than 53 MWd/kg was determined to be ductile (Reference 4.1.5.a-5).

The observations described above indicate that at a burnup level of 60 MWd/kg, the cladding material has [

] a strain-limited cladding failure is not expected at a burnup level of 60 MWd/kg due to an operational transient. Additional confirmation of acceptable cladding performance to rod average burnups up to [

] Acceptable cladding performance to rod average burnups up to approximately 58 MWD/kg was also recently demonstrated for 16x16 fuel assembly designs (ANO-2) (Reference 4.1.5.a-19).

FIGURE 4.1.5.A-1

YIELD STRENGTH AS A FUNCTION OF FLUENCE FOR [ ]

IRRADIATION TEMPERATURE 500 TO 650°F, ELEVATED TEMPERATURE TEST

YIELD STRENGTH, KSI

NEUTRON FLUENCE, n/cm^2 (E > 1 MeV)

ULTIMATE TENSILE STRESS, k psi

TEST TEMPERATURE, °C

FIGURE 4.1.5.A-2
ULTIMATE TENSILE STRENGTH OF SHORT-TRANSVERSE
SPECIMENS IRRADIATED TO 4.3 X 10^{19} N/CM^2 (E > 1 MeV)

FIGURE 4.1.5.A-3

UNIFORM ELONGATION AS A FUNCTION OF FLUENCE FOR [

] ZIRCALOY, IRRADIATION TEMPS. 560-610°F

UNIFORM ELONGATION, %

NEUTRON FLUENCE x 10^{21}, n/cm^2 (E > 1 MeV)

REDUCTION OF AREA, %

TEST TEMPERATURE, °C

FIGURE 4.1.5.A-4

PERCENT REDUCTION OF AREA FOR SHORT-TRANSVERSE SPECIMENS IRRADIATED TO 4.3×10^{19} N/CM^2 (E > 1 MeV)

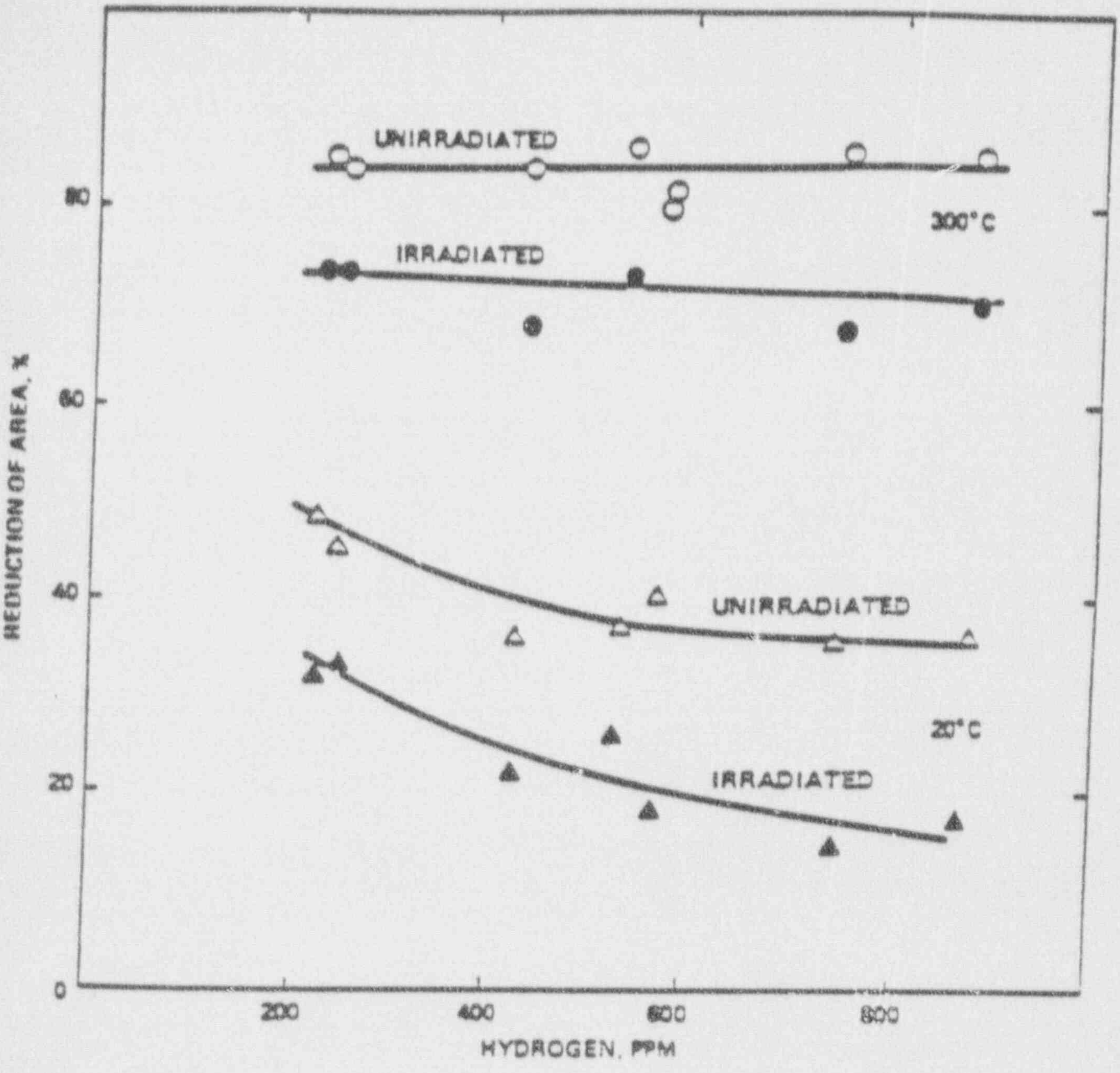


FIGURE 4.1.5.A-5
EFFECT OF HYDROGEN CONCENTRATION ON THE REDUCTION OF AREA FOR ZIRCALOY-2 IRRADIATED TO 10^{20} N/CM^2

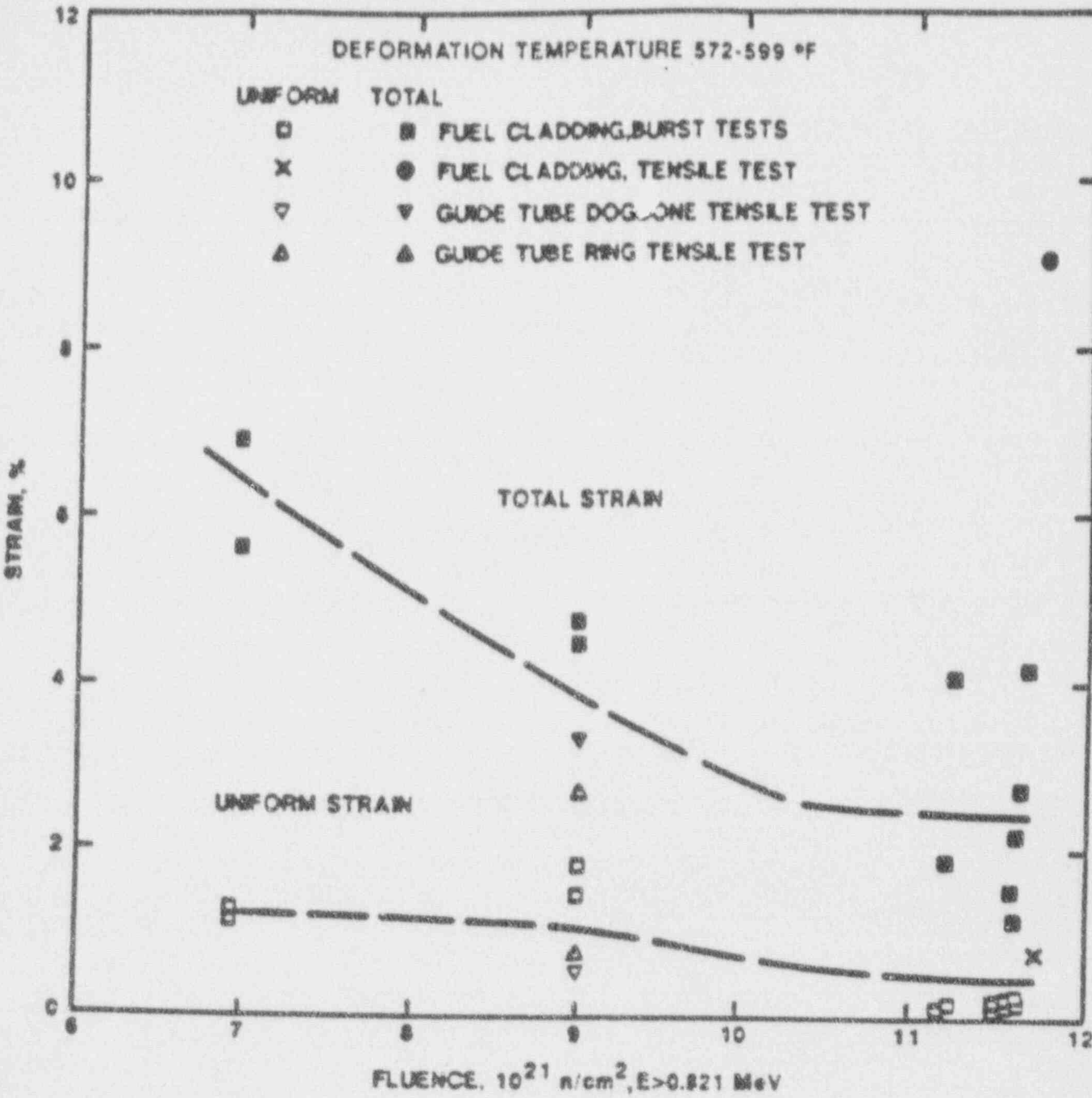


FIGURE 4.1.5.A-6
FLUENCE DEPENDENCE OF STRAIN FOR IRRADIATED ZIRCALOY-4

4.1.6.a Fission Gas Release

The following section supplements Section 4.1.6 of Reference 1.

4.1.6.1.a Fission Gas Release

The calculation of fission gas release is an integral part of the fuel performance calculations involving the temperature distribution and internal pressure of fuel rods. The release of fission product gases plays an important role in the calculation of gas conductivity and, therefore, affects the transfer of heat from the UO_2 pellets to the cladding. C-E's current model for these calculations (FATES3B) was submitted to the NRC in 1986 (Reference 4.1.6.a-1) and received NRC approval in early 1987 (Reference 4.1.6.a-2) . The FATES3B fission gas release model was developed utilizing data from low and high power rods with burnups ranging from 6.5 to 61.5 MWD/kg and measured releases of 0.3 to 48.1%. The model includes the results of fission gas release measurements performed on test rodlets that were irradiated in a PWR and subsequently ramped to high linear heat rates. Comparisons between measurements and FATES3B predictions are given in Reference 4.1.6.a-1.

Additional extended burnup data on fission gas release has been obtained since the publication of References 1 and 4.1.6.a-1. These data consist of six-cycle Fort Calhoun (49.7 to 55.7 MWD/kg) rods and five-cycle Zion-1 rods (54.3 to 59.4 MWD/kg) (References 4.1.6.a-3 and -4). All of these fission gas release measurements were low (less than 2.8% at burnups up to 59.4 MWD/kg). These data also show no significant enhancement of fission gas release with burnup at normal operating levels. These data are presented in Tables 4.1.6.a-1 and 4.1.6.a-2.

Microstructural examinations of the Fort Calhoun rods showed the formation of a porous rim (75 to 80% TD), 150-250 microns thick (References 4.1.6.a-3 and -5). This porous rim can result in a decrease in local fuel thermal conductivity and thus an increase in pellet temperature. C-E believes that this porous layer is a phenomenon associated with local burnup and is well behaved. [

] This increase is not considered significant in low power, high burnup fuel. In addition, other high burnup effects are known to offset the temperature increase due to the porous rim. Two such important effects are [

] Thus, it is concluded that the effects of a porous rim can be neglected for burnups of up to 60 MWD/kg.

High Burnup Data Comparisons

The predictive capability of the FATES3B fuel performance code was demonstrated with respect to fission gas release by comparing code predictions with experimentally measured data in Reference 4.1.6.a-1. The high-burnup data sets (at and above 50 MWD/kg rod average burnup) analyzed as part of the FATES3B correlation and verification data bases were characteristic of fission gas release data in the high-burnup and high-temperature regime. Additional extended burnup data on fission gas released by test rods (typical of fuel rods operated in C-E designed commercial reactors) have been obtained since the publication of Reference 4.1.6.a-1. Comparisons of these data to FATES3B predictions are presented in Table 4.1.6.a-3. These data comparisons provide additional support for FATES3B fission gas release predictions in the high-burnup, low-temperature (low power) regime. These data are described below.

Calvert Cliffs Data:

High-burnup performance evaluations of Zircaloy-4 clad test fuel rods and "all Zircaloy" fuel assemblies were performed on fuel irradiated in Calvert Cliffs 1. The evaluations were sponsored by Combustion Engineering in conjunction with the Electric Power Research Institute (EPRI) (Reference 4.1.6.a-6). A total of 60 test fuel rods were fabricated for this experiment and were equally distributed among three reconstitutable Batch B assemblies. Fission gas release data comparisons were performed for 16 of these test rods, with end of life rod average burnups ranging from 18.7 to 44.4 MWD/kgU, in support of the FATES3B verification effort (Reference 4.1.6.a-1). Five of the modern design test rods, prepressurized

rods with modern design non-densifying pellets, were irradiated one additional (fifth) cycle to burnups of 49.4 to 54.1 MWD/kgU. The fission gas released by the fuel in these rods was measured. A comparison of the measured gas releases with FATES3B predicted gas releases for these five test rods is presented in Table 4.1.6.a-3. On the average, FATES3B [ ]

Fort Calhoun Data:

The Fort Calhoun extended burnup demonstration program was sponsored by the Department of Energy (DOE) to demonstrate the performance of C-E's standard 14x14 fuel design at extended burnups (Reference 4.1.6.a-7). Hot cell examination work on some of the test rods irradiated through six cycles was performed in a follow-on program jointly sponsored by DOE, the C-E Owners Group, and C-E (Reference 4.1.6.a-3). Fission gas release data comparisons were performed for four of the most highly burned test rods (54.6 to 55.7 MWD/kg rod average burnup). These four rods resided in positions very close to each other in the same quadrant of Assembly D005 through the entire irradiation period. A single FATES3B case was generated using design input parameters and an irradiation history that appropriately models all four test rods. The comparisons of measured gas released and the FATES3B predicted gas release are also presented in Table 4.1.6.a-3. On the average, FATES3B [ ]

Conclusions:

Additional data comparisons have been performed on fuel rods typical of C-E current generation fuel that were irradiated under normal low-temperature conditions during extended burnup operation to rod average burnups of up to 55.7 MWD/kg. In general, the low temperature release due to knock-out and recoil is [ ] at 60 MWD/kg. However, releases associated with knock-out and recoil are low. Therefore, it can be concluded that FATES3B adequately models, on a best-estimate basis, the fission gas release of extended-burnup fuel operated under normal conditions in C-E designed commercial reactors.

4.1.6.2.a <u>Evaluation of Fission Gas Release</u>

The discussion in Section 4.1.6.1.a surveys the situation at C-E with respect to the data available and the modeling of the fission gas release of fuel burned to extended burnups. Significant strides have been achieved in the area of normal operation and in the area of response to ramps. The conclusions are:

(1) Commercial fuel rods operating in PWRs with helium prepressurization and nondensifying fuel have been examined and consistently found to contain very low levels of released fission gases to burnup levels of ~60 MWD/kg. The relative absence of significant enhancement due to burnup at normal operating levels is now verified by direct measurement.

(2) Data evaluated by C-E support the FATES3B model to burnups of 60 MWD/kg. Furthermore, the trends observed in all UO_2 behaviors are gradual and support the orderly extension of the allowable burnup.

Table 4.1.6.a-1

FISSION-GAS RELEASE DATA FROM FORT CALHOUN FUEL RODS

Rod Number	Burnup, MWD/MTU	Rod Time-Avg. Heat Rating, (kw/ft)	Total Gas Collected, cc STP	Volume (Xe+Kr) Released, cc STP	Volume (Xe+Kr) Generated, cc STP(a)	% Fission Gas Released
KJD008	51500	5.38	794.1	23.4	3425.1	0.68
KJD015	51400	4.98	769.3	19.2	3417.4	0.56
KJE051	55700	5.36	753.8	46.5	3704.7	1.26
KJE077	55400	5.39	736.5	49.1	3684.6	1.33
KJE052	54600	5.25	767.6	33.0	3633.1	0.91
KJE006	49700	5.23	737.8	31.6	3309.3	0.95
KJD072	53400	5.12	755.2	22.1	354[illegible].6	0.62
KJD075	51500	(b)	747.1	22.9	3429.3	0.67
KJE109	52900	5.26	735.8	46.1	3517.2	1.31
KJE068	52600	(b)	[illegible]51.7	27.4	3500.9	0.78
KJE089	53100	5.50	745.0	40.7	3534.7	1.15
KJE088	52900	5.45	730.5	36.6	3516.6	1.04

(a) Assumes production rate of 30 atoms of (Xe+Kr) per 100 fissions and 200 MeV/fission.

(b) Physics Data used to calculate the time-average heat rating are not available for these rods.

Table 4.1.6.a-2
FISSION-GAS RELEASE DATA FROM ZION 1 FUEL RODS

Rod No.	Rod Avg. Burnup (MWD/kg)	Total Gas Collected (cc at STP)	Xe + Kr Release (cc at STP)	Xe + Kr Generated (a) (cc at STP)	Fission Gas Release (%)	Kr/Xe Ratio
A10	55.90	795.25	63.30	3711.91	1.71	0.098
G12	58.25	778.23	91.29	3867.62	2.36	0.107
612	55.34	779.65	43.74	3674.15	1.19	0.115
613	57.99	811.62	70.61	3854.77	1.83	0.121
618	55.32	773.01	42.98	3672.56	1.17	0.097
620	55.29	772.67	45.28	3658.88	1.24	0.087
638	54.26	783.13	51.76	3606.58	1.44	0.111
640	56.06	777.38	48.20	3737.39	1.29	0.099
648	56.60	773.73	43.25	3774.12	1.15	0.116
650	58.80	851.89	105.72	3903.94	2.71	0.128
653	59.43	804.34	90.01	3950.02	2.28	0.108
657	54.78	767.56	38.61	3636.84	1.06	0.105
659	56.23	764.05	54.09	3721.29	1.45	0.099
665	55.78	785.46	52.47	3707.75	1.42	0.123
679	56.62	744.28	49.94	3771.20	1.32	0.088
681	55.01	768.30	41.80	3675.09	1.14	0.103
683	56.32	754.43	50.17	3762.88	1.33	0.099
685	55.35	766.59	44.08	3667.13	1.20	0.093
693	55.98	782.32	41.38	3704.94	1.12	0.086
696	58.13	816.72	73.75	3863.88	1.91	0.116
697	56.37	738.07	38.08	3727.14	1.02	0.112

(a) Assumptions: Fission Gas Yield = 0.3 atoms (Xe + Kr) per fission
Energy Release = 200 MeV/fission

Table 4.1.6.a-3

FATES3B Predictions of Gas Release from High Burnup, Low Power Test Rods

Rod	Rod Average Burnup MWD/kg	Measured Gas Release	Predicted Gas Release %	Predicted-Measured Gas Release %
Calvert Cliffs-1				
SN24	49.4	1.16		
SN34	49.4	0.67		
SN36	49.5	1.00		
SN45	54.1	>2.02		
SN59	49.7	1.03		
Fort Calhoun Extended Burnup Fuel				
KJE051	55.7	1.26		
KJE052	54.6	0.91		
KJE077	55.4	1.33		
KJE109	54.6	1.31		

4.1.7.a Fuel Thermal Conductivity

The following paragraph appends Subsection 4.1.7.3 of Reference 1.

4.1.7.3.a Evaluation of Fuel Thermal Conductivity

No new data on the thermal conductivity of irradiated fuel has become available since the publication of Reference 1. However, the performance of fuel rods to 60 MWD/kg (References 4.1.7.a-1 and -2) indicates no trend toward serious degradation of thermal conductivity. The ability of the FATES3B model to predict the measured gas release data suggests that any degradation in local fuel thermal conductivity, such as due to the formation of a porous rim, is implicitly accounted for in the FATES3B model. This is thought to be accomplished by the density correction in the fuel thermal conductivity equation and through the conservatism that exists in the other parts of the relevant submodels used in the fission gas release calculation. It is therefore concluded that the current thermal conductivity equations are adequate to 60 MWD/kg.

4.1.8.a Fuel Melting Temperature

The following paragraph appends Subsection 4.1.8.1 of Reference 1.

4.1.8.1.a Modeling of Fuel Melting Temperature and Effect of Increased Burnup

New data continue to support the conservatism of the melting point expression. The range of the melting point determinations of unirradiated UO_2 fabricated by C-E (5094-5173°F) performed at Pacific Northwest Labs (Reference 4.1.8.a-1) exceeds the melting point calculated by the expression for unirradiated fuel (5080°F). Work reported by Komatsu, et al, (Reference 4.1.8.a-2) showed no effect of burnup on UO_2 irradiated up to burnups of 30 MWD/kg, and a drop of only ~2°F/MWD/kg for UO_2-20%PuO_2 irradiated up to burnups of 110 MWD/kg. Thus, it is concluded that the melting point expression is adequate to 60 MWD/kg.

4.1.9.a <u>Fuel Swelling</u>

The following paragraph appends Subsection 4.1.9.3 of Reference 1.

4.1.9.3.a <u>Evaluation of Fuel Swelling</u>

Data for six-cycle fuel rods from Fort Calhoun and five-cycle fuel rods from Zion 1 (References 4.1.9.a-1 and -2, respectively) have become available since the publication of Reference 1. Fuel density measurements were made on pellet sections with a local burnup of 60.4 MWD/kg from Zion 1 and 63.3 MWD/kg from Fort Calhoun. These data and lower burnup data from previous cycles of these reactors indicate a swelling rate of 0.53%/10 MWD/kg for Zion 1 and 0.70%/10 MWD/kg for Fort Calhoun, which is entirely consistent with the 0.4-0.8%/10 MWD/kg data measured previously for Fort Calhoun and Calvert Cliffs-1. These results show no enhancement of the fuel swelling rate for local fuel burnups up to 63.3 MWD/kg, indicating no change in the fuel swelling mechanism up to this burnup level. Consequently, the current FATES3B model is valid to these high burnups.

4.1.10.a Fuel Rod Bow

The discussion provided in Section 4.1.10 of Reference 1 applies to the proposed increase in the 1-pin burnup limit to 60 MWD/kg. Rod bow is not a concern for high burnup fuel rods since their power falloff more than compensates for their rod bow penalty.

4.1.11.a Fretting Wear

The discussion provided in Section 4.1.11 of Reference 1 applies to the proposed increase in the 1-pin burnup limit to 60 MWD/kg. No significant fretting wear has been seen during extensive inspections of C-E fuel rods and the degree of stress relaxation of the grid springs and creepdown of the fuel rod changes very little after one operating cycle.

4.1.12.a Pellet/Cladding Interaction

The following section replaces Section 4.1.12 of Reference 1.

C-E has been invclved in many ramping experiments and has collected a considerable amount of PCI data. The data plotted in Figure 4.1.12.a-1 came from rodlets pre-irradiated at Obrigheim and ramped at either the Petten or Studsvik test facilities in Europe (References 4.1.12.a-1, -2, -3, -4). The data shown are only from rodlets using the standard C-E or KWU designs. Other data available in the literature have not been shown because of design differences. It is important to recognize that comparisons between experimental PCI results are only valid when the important design variables are consistent. All of these rodlets were preconditioned in a PWR at similar power levels and were ramped under PWR conditions at relatively fast and consistent rates (50-110 W/cm/min). Data are also available at slower ramp rates. The slower ramps are less severe and give improved PCI performance. [

] In addition, as burnup increases, the capability of the fuel to reach the power levels needed for PCI failure is diminished. This fact[

]

FIGURE 4.1.12.a-1
PCI TEST RESULTS ON STANDARD C-E AND KWU RODLETS

LINEAR HEAT RATING, KW/FT

BURNUP, MWD/KG

4.1.13.a <u>Cladding Deformation & Rupture</u>

The discussion provided in Section 4.1.13 of Reference 1 applies to the proposed increase in the 1-pin burnup limit to 60 MWD/kg. It has been determined that the LOCA models for cladding deformation and rupture are adequate for use at 60 MWD/kg. [

]

4.1.14.a Fuel Rod Growth

The following replaces Subsection 4.1.14 of Reference 1.

It has been well established that Zircaloy-4 clad rods exhibit axial elongation or growth when continuously exposed to a neutron flux. A substantial amount of growth data has been obtained on PWR fuel rods of modern design (i.e., pressurized rods with nondensifying fuel) at burnups [ ]. This information has been used to modify the fuel rod growth models originally developed with data obtained at lower fluences and from rods of older design (densifying fuel with lower initial pressurization levels).

4.1.14.1.a Modeling of Fuel Rod Growth

The overall elongation of a Zircaloy clad fuel rod is due to several contributing mechanisms including stress-free irradiation growth of the Zircaloy cladding, mechanical interaction between the UO_2 fuel pellets and the Zircaloy cladding, and a net positive growth component due to creepdown of the cladding under the external coolant pressure. Each of these contributing mechanisms are related to the time of operation through accumulated burnup or fluence. Rather than account for individual contributions from each mechanism, overall fuel rod growth is measured and empirically modeled for design purposes.

Growth strain versus fluence (E>0.821 MeV) is linear on a log-log plot. The functional form of such an equation is:

$$\epsilon = A\,(\phi t)^n$$

where ϵ = strain, percent.

ϕt = neutron fluence, n/cm^2 (E>0.821 MeV) x 10^{-21}

A and n = constants, as shown below.

A regression analysis was used to determine the value of the constants A and n and resulted in the following growth equations:

Upper 95% tolerance limit: ϵ = [

Best estimate equation: ϵ =

Lower 95% tolerance limit: ϵ =]

The growth data used in this analysis covered a fluence range of [ ].

4.1.14.2.a <u>Effect of Extended Burnup</u>

Rod length measurements performed on rods with fast fluences up to [ ] have shown continuous and well-behaved growth with increasing exposure (References 4.1.14.a-1 through 4.1.14.a-8). These data have confirmed that no acceleration of the growth rate or other abrupt changes occur up to the exposure levels of the examined rods. Furthermore, fuel rod growth at higher burnups appears to be relatively insensitive to slight design differences. [

] do not contribute as much to the overall growth rate at higher exposures as would be inferred from measurements taken after only one or two operating cycles. This observation is supported by measurements taken as part of fuel performance evaluation programs at Fort Calhoun, Calvert Cliffs-1, and Arkansas Nuclear One-Unit 2 (References 4.1.14.a-3, -5, -6, -7, -8).

4.1.14.3.a <u>Evaluation of Fuel Rod Growth</u>

Figure 4.1.14.a-1 shows growth measurements obtained on C-E fuel rods compared to the C-E fuel rod growth model described in Subsection 4.1.14.1.a. Data from 14x14 fuel rods at Calvert Cliffs-1 and Fort Calhoun have been obtained for fluences of up to [ ] (References 4.1.14.a-6, -8) while data from 16x16 fuel rods at Arkansas Nuclear One-

Unit 2 have been obtained to fluences of [ ] (Reference 4.1.14.a-7).

The growth data from the Calvert Cliffs-1 fuel rods have also been used in an analysis of growth published by Franklin which involved more than 700 fuel rod length measurements (Reference 4.1.14.a-9). This analysis confirmed the well-behaved nature of fuel rod growth at high fluence and [ ].

The database shown in Figure 4.1.14.a-1 includes measurements from ANO-2 fuel rods that showed higher growth than other rods in the same batch. The higher growths are believed to be related to the relatively high carbon content of the cladding. A similar association between the carbon content of cladding and fuel rod growth was also reported in the 1988 ANS Topical Meeting on LWR Fuel Performance by Fragema, describing performance of fuel rods irradiated in TN1. [

]

Figure 4.1.14.a-1

FUEL ROD GROWTH MEASUREMENTS
COMPARED TO C-E ZIRCALOY FUEL ROD GROWTH MODEL

GROWTH STRAIN, % Δ IN./IN.

FLUENCE X E-21 n/cm^2, (E>0.821 MeV)

4.2.1.a <u>Guide Tube Wear</u>

The discussion provided in Section 4.2.1 of Reference 1 applies to the proposed increase in the 1-pin burnup limit to 60 MWD/kg. An extensive program was initiated in response to the detection of guide tube wear. This program resulted in (a) the development of a guide tube wear sleeve design that essentially eliminates the concern of guide tube wear (Reference 4.2.1.a-1), and (b) the development of an unsleeved fuel assembly design that reduces guide tube wear to acceptable levels.

The only unsleeved designs being fabricated are the fuel assemblies for the System 80 reactors. Post-irradiation examinations of these assemblies (Reference 4.2.1.a-2 and Reference 4.2.1.a-3) have verified the conservatism of the analytical predictions used to justify the unsleeved assemblies. As discussed in Reference 1, the defense of the unsleeved assembly design for extended burnup operation [

]

4.2.2.a Fuel Assembly Length Change and Shoulder Gap Change

This section replaces Section 4.2.2 of Reference 1 in its entirety.

Fuel assembly length change results from two distinct mechanisms in the Zircaloy guide tubes: irradiation induced growth and compressive creep. Growth is produced by radiation effects on the Zircaloy crystalline structure, and causes the guide tubes to elongate. Compressive creep is the permanent reduction in length of the guide tubes in response to net holddown force on the fuel assembly structure.

Change in guide tube length affects the fuel assembly engagement with the reactor internals (thereby affecting the holddown force on the assembly) and the shoulder gap (the distance between the top of the fuel rods and the bottom of the upper end fitting). The length change is important in the evaluation of criteria pertaining to each of these aspects of fuel performance.

Since the holddown force is a function of fuel assembly length, irradiation induced guide tube growth causes an additional compression of the upper end fitting springs, increasing the compressive load on the guide tubes. The higher load in turn causes an increased compressive creep rate of the guide tubes. Therefore, the net fuel assembly length change at a given time during operation requires a time history analysis to properly account for the combined effects of irradiation growth and creep up to that point in time.

4.2.2.1.a Modeling of Assembly Length Change and Shoulder Gap Change

a) Assembly Length Change

Growth and creep characteristics are dependent on the metallurgical state of the Zircaloy guide tubes. The analytical models presented in Reference 1 for [ ] have

been updated, based on all the available guide tube length change data on C-E fuel assemblies with [   ]. The general forms of the equations presented in Reference 1 for the irradiation induced growth model and the axial creep model were maintained while the proportionality factors and exponential constants were adjusted to obtain a best fit of the data. Uncertainties on the guide tube length change predictions were based on an evaluation of the errors between measured data and best estimate predictions. The result was a fluence dependency of the length change uncertainty. The updated irradiation induced growth model and axial creep model for [   ] are summarized in Table 4.2.2.a-1, along with the uncertainty function on the guide tube length change.

Dimensional changes of fuel assembly guide tubes are analytically predicted by the SIGREEP computer code, which is described in Reference 4.2.2.a-1. The code utilizes a computerized Monte Carlo technique for establishing resultant joint probability density functions by randomly selecting combinations of input values to be used in a time history analysis of dimensional changes. Inputs assigned statistical uncertainties include component dimensions, the assembly uplift force and the probability/confidence factor of the guide tube length change model (see Item 4 of Table 4.2.2.a-1).

The SIGREEP computor code generates a set of randomly selected values for the input parameters that have been assigned uncertainty distributions, and then uses that set of inputs to perform a time history analysis of the guide tube length change. When the analysis reaches the specified operating time or burnup, the dimensional change prediction for the fuel assembly is complete. A single value of assembly length change is the result of the time history calculation. The same steps are repeated (starting with a different set of randomly selected values for the input parameters) until a sufficient number of cases (typically 2000) have been generated to define a probability histogram of length change at end of life (EOL). The resultant histogram represents the statistical variation of EOL length change which can be attributed to the uncertainties of the input parameters. Values can be chosen from the histogram at desired probability levels for comparisons to

actual data or appropriate design criteria. Figure 4.2.2.a-1 presents a typical histogram of fuel assembly length change.

b) Shoulder Gap Change

Shoulder gaps change with residence time in the reactor due to differential growth between the fuel rods and the fuel assembly structure (guide tubes). Reference 1 described a technique of evaluating shoulder gap change using the SIGREEP computer code. With that technique, fuel assembly length change is calculated by SIGREEP exactly as described above, but for each time history case for fuel assembly length change, fuel rod length is simultaneously calculated using values for the growth coefficient and beginning of life (BOL) dimensions that have been randomly selected from the probability distributions for these parameters. When the time history case reaches the specified time or burnup, shoulder gap change is calculated as the difference in fuel rod and fuel assembly length changes. A single value of shoulder gap change is the end product of the time history calculation. The calculation is repeated with different sets of randomly selected values for the input parameters until a sufficient number of cases (again typically 2000) have been generated to define a probability histogram of shoulder gap at EOL.

This method of evaluating shoulder gap change is used on 14x14 fuel designs but, because of the high fuel rod growth rate associated with some ANO-2 Batch C fuel rods, an interim approach of deterministically combining a conservatively high fuel rod growth prediction with a conservatively low fuel assembly growth prediction had been used on 16x16 fuel designs, pending more 16x16 measurement data. Additional fuel rod growth data are now available and are presented in Section 4.1.14.a, along with an updated fuel rod growth model based on the data. Also included in Section 4.1.14.a is a discussion of the cause of the high growth rates of the ANO-2 Batch C rods and a justification for no longer applying those high growth rates to current fuel designs (i.e., a change in the material specification of the cladding). The interim approach is, therefore, no longer necessary and the shoulder gap evaluation technique utilizing the SIGREEP computer code with the updated fuel rod growth model of Section 4.1.14.a can be used for 16x16 fuel designs.

A comparison of this technique to shoulder gap measurements taken on 16x16 fuel assemblies with [ ] is included in Section 4.2.2.4.a.

4.2.2.2.a Effect of Extended Burnup

As stated in the preceding sections, fuel assembly length change is the net change resulting from irradiation induced growth and compressive creep of the guide tubes. Since growth is fluence dependent and compressive creep is time and flux dependent, assembly length change and shoulder gap are affected by extended burnup. In general, higher burnups are expected to result in greater increases in assembly length, greater holddown spring compression, and larger changes in shoulder gap. The extent of these changes will be evaluated based on the specific extended burnup operating conditions and the particular fuel assembly design.

4.2.2.3.a Evaluation of Assembly Length Change

Guide tube length change data for C-E fuel assemblies with [ ] are shown in Figures 4.2.2.a-2 thru 4.2.2.a-5, along with SIGREEP predictions using the irradiation induced growth equation and axial creep equation from Table 4.2.2.a-1. Figures 4.2.2.a-2 thru 4.2.2.a-5 present data and predictions for fuel designs that have different axial loads on the fuel assembly. The different axial loads result from differences in holddown spring forces and/or uplift forces on the fuel assembly spacer grids. These differences affect the axial creep component of the guide tube length change so the various sets of data must be presented on separate figures.

Inspection of the four figures shows that the best estimate SIGREEP predictions are in good agreement with the data, both in the magnitude of the predictions and the trend of the predictions. In addition, the upper and lower 95% predictions represent conservative estimates of the guide tube length changes. The good agreement between the data and the SIGREEP predictions in all four figures confirms the creep model's correct

sensitivity to the axial stress on the guide tubes. Based on the comparisons of the data and the predictions, it is concluded that both the analytical model and the growth and creep equations are acceptable for use in predicting fuel assembly length change for designs with [ ] to extended burnups.

4.2.2.4.a Evaluation of Shoulder Gap Change

Shoulder gap change data for C-E fuel assemblies with [ ] are shown in Figures 4.2.2.a-6 thru 4.2.2.a-8 along with the limiting shoulder gap change prediction using the technique described in Section 4.2.2.1.a. The SIGREEP predictions shown on the figures were generated using a typical ratio between the fuel rod fluence and the guide tube fluence. Three separate figures are provided because the fuel assembly designs associated with each figure have different guide tube length change characteristics (see above section) which, in turn, affect the shoulder gap change characteristics.

Inspection of the three figures shows that the analytical predictions represent conservative bounds of the data. Shoulder gap change data for fuel assemblies with [ ] are available to fluences of approximately [ ] nvt. Fuel rod growth data exist to considerably higher fluences (over [ ] nvt, per Figure 4.1.4.a-1) and were included in the development of the revised fuel rod growth model. Since the fuel rod growth is the predominant component in the shoulder gap change and the technique of predicting the limiting shoulder gap change employs the fuel rod growth equation that properly models the high fluence rod data, the fuel rod growth portion of the shoulder gap change analysis is acceptable for use to extended burnups. The guide tube growth portion of the shoulder gap change analysis uses the model verified to extended burnups in the above section. Therefore, it is concluded that the analytical technique for predicting shoulder gap changes (SIGREEP) can be used to conservatively predict shoulder gap changes for designs with [ ] to extended burnups.

Table 4.2.2.a-1

Analytical Models for [ ]

1. Overall Length Change Model

Length Change = Irradiation Growth - Compressive Creep ± Uncertainty

2. Irradiation Growth Model

Equation Form: $\epsilon = A\,(\phi t)^n$

where: ϵ = axial growth strain, in/in

A = proportionality factor = [ ]

ϕt = fluence, nvt x 10^{-21} (E > 0.821 MeV)

n = exponential constant = [ ]

3. Compressive Creep Model

Equation Form: $\epsilon = \alpha\,\beta\,(\sigma)^N$

where: ϵ = axial creep strain rate, in/in/hr

α = proportionality factor = [ ]

$\beta = (\phi)^{0.85}\, e^{(-6000/RT)}\, (Ake^{kt} + C)$

ϕ = fast neutron flux, n/cm^2-sec (E > 1.0 MeV)

R = 1.987 cal/mol °K

T = temperature, °K

A = constant = [ ]

t = time, hrs

k = constant = [ ]

C = constant = [ ]

σ = axial guide tube stress, ksi

N = exponential constant = [ ]

4. Uncertainty Model

Equation Form: Uncertainty = KS

where: K = probability/confidence factor e.g. 1.96 for 95/95

S = standard deviation = [ ]

ϕt = fluence, nvt x 10^{-21} (E > 0.821 MeV)

FIGURE 4.2.2.a - 1

TYPICAL PROBABILITY HISTOGRAM FOR FUEL ASSEMBLY LENGTH CHANGE

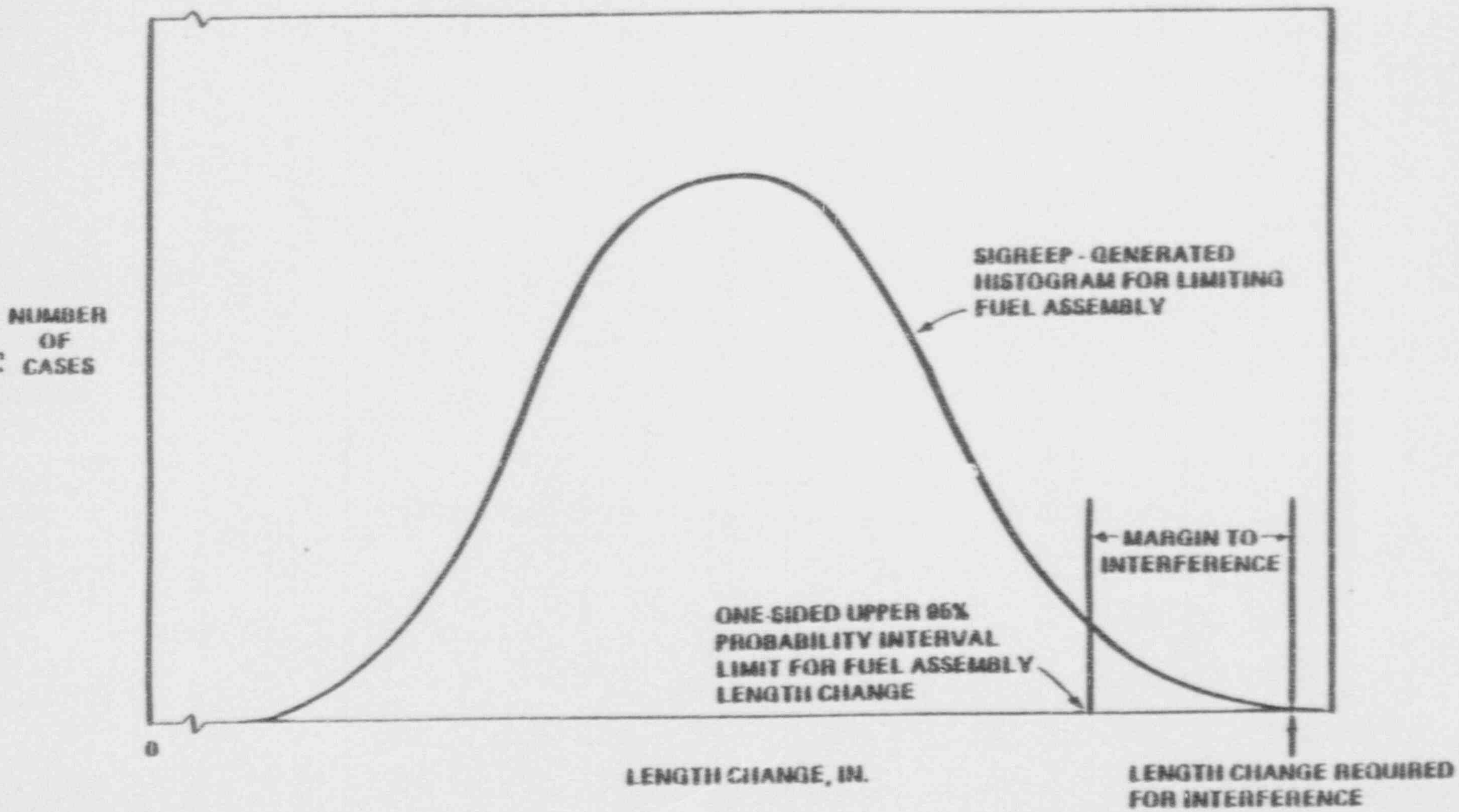


FIGURE 4.2.2.a-2

COMPARISON OF GUIDE TUBE LENGTH CHANGE DATA TO
SIGREEP PREDICTIONS FOR ANO-2 FUEL ASSEMBLIES

GUIDE TUBE LENGTH CHANGE, IN.

GUIDE TUBE FLUENCE, NVT X 1.0E-21 (E>0.821 MeV)

FIGURE 4.2.2.a-3

COMPARISON OF GUIDE TUBE LENGTH CHANGE DATA TO SIGREEP PREDICTIONS FOR SONGS FUEL ASSEMBLIES

GUIDE TUBE LENGTH CHANGE, IN.

GUIDE TUBE FLUENCE, NVT X 1.0E-21 (E>0.821 MeV)

FIGURE 4.2.2.a-4

COMPARISON OF GUIDE TUBE LENGTH CHANGE DATA TO SIGREEP PREDICTIONS FOR PVNGS FUEL ASSEMBLIES

GUIDE TUBE LENGTH CHANGE, IN.

GUIDE TUBE FLUENCE, NVT X 1.0E-21 (E>0.821 MeV)

FIGURE 4.2.2.a-5

COMPARISON OF GUIDE TUBE LENGTH CHANGE DATA TO SIGREEP PREDICTIONS FOR ST. LUCIE 2 FUEL ASSEMBLIES

GUIDE TUBE LENGTH CHANGE, IN.

GUIDE TUBE FLUENCE, NVT X 1.0E-21 (E>0.821 MeV)

FIGURE 4.2.2.a-6

COMPARISON OF SHOULDER GAP CHANGE DATA TO

SIGREEP PREDICTIONS FOR ANO-2 FUEL ASSEMBLIES

SHOULDER GAP DECREASE, IN.

FUEL ROD FLUENCE, NVT X 1.0E-21

FIGURE 4.2.2.a-7

COMPARISON OF SHOULDER GAP CHANGE DATA TO

SIGREEP PREDICTIONS FOR SONGS FUEL ASSEMBLIES

SHOULDER GAP DECREASE, IN.

FUEL ROD FLUENCE, NVT x 1.0E-21

FIGURE 4.2.2.a-8

COMPARISON OF SHOULDER GAP CHANGE DATA TO

SIGREEP PREDICTIONS FOR PVNGS FUEL ASSEMBLIES

SHOULDER GAP DECREASE, IN.

FUEL ROD FLUENCE, NVT X 1.0E-21

4.2.3.a Fuel Assembly Holddown

The discussion provided in Section 4.2.3 of Reference 1 applies to the proposed increase in the 1-pin burnup limit to 60 MWD/kg. The holddown spring relaxation due to extended burnup tends to be offset by concurrent growth of the fuel assembly.

4.2.4.a Grid Irradiation Growth

The discussion provided in Section 4.2.4 of Reference 1 applies to the proposed increase in the 1-pin burnup limit to 60 MWD/kg. Since the grid growth data presented in Reference 1 agreed well with all the other growth measurements [ ] presented in that reference, the good agreement between the growth measurements and predictions for [ ] presented in Reference 1 supports the adequacy of the grid irradiation growth model to extended burnup.

4.2.5.a <u>Spacer Grid Relaxation</u>

The discussion provided in Section 4.1.13 of Reference 1 applies to the proposed increase in the 1-pin burnup limit to 60 MWD/kg. The degree of stress relaxation of the grid springs and creepdown of the fuel rod changes very little after one operating cycle. Also, the observation of superior performance of the grids in the extended burnup demonstration assemblies irradiated in Calvert Cliffs Unit 1 and ANO-2 confirm that the relaxation of the fuel assembly spacer grid springs is not a concern for the extended burnup operation of 14x14 or 16x16 fuel assembly designs.

4.2.6.a Corrosion of the Fuel Assembly Structure

The following paragraphs append Subsection 4.2.6.3 of Reference 1.

4.2.6.3.a Evaluation of Corrosion of the Fuel Assembly Structure

Additional in-reactor corrosion data will be obtained from hot cell examinations (metallographic and hydrogen content analyses) to be performed on a five-cycle Calvert Cliffs-1 assembly cage that experienced an assembly average burnup of [ ]. Detailed poolside visual examinations were performed on this assembly. No indications of anomalous behavior, such as oxide spalling or structural cracking, were observed. The hot cell data, which will be available in 1989 or 1990 from a joint EPRI, BG&E and C-E program, are expected to support the current model which predicts the oxide layer thickness to increase monotonically with time.

On review of the available information, it is concluded that, for the coolant conditions typical of ANO-2, the corrosion of the Zircaloy-4 structure will not preclude the operation of C-E 16x16 fuel assemblies to 1-pin burnups of 60 MWD/kg. For reactors with higher coolant temperatures and coolant chemistry conditions differing from ANO-2, such as higher lithium concentrations, further evaluations of the assembly structure corrosion behavior would have to be made.

4.2.7.a Burnable Poison Rod Behavior

The following subsections replace the corresponding subsections of Reference 1.

4.2.7.1.a Modeling of Burnable Poison Rod Behavior

Al_2O_3-B_4C Pellet Swelling. The swelling of the burnable poison material, induced by irradiation, results in dimensional changes which can affect cladding strain and poison rod void volume. The neutron absorber material employed in the poison rods is in a pelletized form and consists of a dispersion of boron carbide (B_4C) particles in an alumina (Al_2O_3) matrix. The B_4C content is established by core neutronic requirements and has ranged to levels on the order of 4 wt%. The dimensional changes of the pellet are predicted by a model which assumes [

].

Since the Al_2O_3 swelling is the dominant contributor to pellet swelling at high exposure, the Al_2O_3-B_4C swelling is related to fast fluence in the model. It is recognized, however, that the swelling of B_4C is a function of thermal flux to the extent that it depends upon the B-10 (n,α) Li-7 reaction.

In relating pellet swelling to irradiation exposure, it is assumed [

]. The B_4C swelling rate used is the same as in C-E's model for B_4C swelling in a control element assembly (CEA) as described in Reference 4.2.7.a-1, i.e., a volumetric swelling of 0.3% per percent B-10 burnup. The Al_2O_3 swelling behavior is based on the data reported by Keilholtz and Moore for high density (> 99% TD) pellets (Reference 4.2.7.a-2). Since Al_2O_3 swelling is caused by fast neutron irradiation damage, Keilholtz and Moore correlated their observed Al_2O_3

volume increases with fast fluence (E > 1 MeV).

A review of the data reported by Keilholtz and Moore (Reference 4.2.7.a-2) indicates that for gross overall dimensional changes, a two-stage swelling rate model is an appropriate representation for Al_2O_3 swelling. That is, above a fast fluence of approximately 2.6 x 10^{21} n/cm^2, the swelling of Al_2O_3 is enhanced by microcracking and grain boundary separation which causes a sharp increase in the apparent overall swelling rate. This enhancement of swelling was incorporated into the previous model which was described in Reference 1. However, since the volume created by microcracking accommodates the gas inventory in the rod, this enhancement of swelling does not reduce the poison rod internal void volume available to the gas inventory. Thus, the more accurate model of void volume reduction due to Al_2O_3 swelling is represented by the following expression that accounts for the matrix swelling of Al_2O_3 only:

[

]

The model assumes that swelling is independent of temperature since poison pellets are not expected to exceed an operating temperature of 500°C in PWR applications. Further, Keilholtz and Moore found no significant temperature dependency for Al_2O_3 swelling in the range of 300 to 600°C.

[

] a two-stage model is used for the composite Al_2O_3-B_4C pellet swelling model. The volumetric swelling rate for B_4C (i.e., 30% at 100% B-10 depletion) was used in conjunction with Equation (1) for Al_2O_3 to arrive at the following expressions for the volumetric swelling of the composite Al_2O_3-B_4C pellet.

The above relationship for swelling as a function of fluence is plotted in Figure 4.2.7.a-1 for Al_2O_3-B_4C with a B_4C content of 3 wt%. Also plotted are volumetric swelling values calculated from diametral swelling data which were obtained in C-E sponsored post-irradiation examination programs to verify the performance of the Al_2O_3 and Al_2O_3-B_4C pellets. These data consist of direct diameter measurements on 42 whole Al_2O_3-B_4C and 16 whole Al_2O_3 pellets which were from poison rods discharged after 1 cycle of exposure. The results of the post-irradiation examination of these 1-cycle Al_2O_3-B_4C pellets substantiated the assumption of isotropic swelling behavior (i.e., equal axial and diametral swelling rates). It was also found that swelling was independent of initial pellet density in the density range of 85 to 98% TD. In addition, indirect diametral swelling data were obtained, at higher exposures, by profilometry measurements on unpressurized burnable poison rods of the early 14x14 design (described in Table 4.2.7.a-1) discharged after 2, 3 and 4 cycles of reactor irradiation. The pellet diametral swelling in these rods was inferred by conservatively assuming that the Zircaloy-4 cladding had crept down to contact the pellets. This approach had the advantage of directly determining the mechanical performance characteristics of interest at high fluence: (1) the

cladding strain as affected by pellet swelling and (2) by inference, the restrained swelling behavior of the Al_2O_3-B_4C pellets. It was found that even after 4 cycles of reactor operation, the average cladding strain was still negative, exhibiting only a slight tendency to be less negative than the 1-cycle value. Moreover, after 4 cycles, the cladding had completely crept down to contact the pellets and conformed to the pellet shapes as shown by the profile traces. The inferred Al_2O_3-B_4C pellet swelling in these rods, shown in Figure 4.2.7.a-1, was calculated from the irradiated diameter profiles, the as-fabricated cladding wall thickness, and the as-fabricated pellet diameter. It should be noted that, because of the different measurement techniques, the 1-cycle pellet data represent an unrestrained condition, while the higher exposure data derived from rod profiles represent a restrained condition.

A comparison of the performance data with the model in Figure 4.2.7.a-1 indicated the following:

- o The swelling of Al_2O_3-B_4C pellets, as well as that of Al_2O_3 pellets, that occurred during the first-cycle of irradiation up to a fluence of about 3.5 x 10^{21} n/cm^2 (E > 1 MeV) are reasonably predicted by Equations (2) and (3). The data scatter indicated that several 1-cycle Al_2O_3-B_4C pellets apparently swelled more than predicted by the model, most likely due to pellet microcracking.

- o There was no measurable diametral swelling of the pellets contained in the early 14x14 design burnable poison rods exposed to additional irradiation up to 4 cycles, equivalent to 8.2 x 10^{21} n/cm^2 (E > 1 MeV). The reason for the lack of apparent diametral swelling is related to the following overall swelling behavior mechanisms:

  (a) B_4C particle swelling caused by the B-10 (n,α) Li-7 reaction induces microcracking and grain boundary separation in the pellet structure.

(b) The resulting early apparent swelling (while the B-10 is depleting) is enhanced by this void contribution when the pellet is not restrained (This may account for any underprediction of 1-cycle swelling).

(c) At higher fluence (i.e., afte 100% B-10 depletion) some of these new voids are accommodating the Al_2O_3 matrix swelling due to cladding restraint. Once the accommodation is completed, diametral swelling, and therefore, volumetric swelling, would proceed at the swelling rate indicated by Equation (3).

The subsections of Gas Release, Poison Rod Growth, and Poison Rod Cladding Creep of Reference 1 apply to the proposed increase in the 1-pin burnup limit to 60 MWD/kg.

Poison Rod Internal Pressure. The internal pressure at operating conditions is predicted by an analysis involving the calculation of the poison rod void volume, gas temperature, and pellet temperature at operating conditions. Each of the conditions discussed above represents either a time-dependent, fluence-dependent, or power-history dependent mechanism which will produce changes in the poison rod internal pressure through changes in the void volume and the amount of helium released.

Calculation of the EOL internal pressure is predicted for appropriate EOL conditions which include the number of moles of helium (prepressure plus gas released from the pellets), gas temperature (the 100% depleted poison pellets produce only a small amount of heat flux due to gamma heating), and the void volume (reflecting changes due to different temperatures, pellet swelling, poison rod growth, and cladding creepdown).

Also, for the extended-burnup reference designs, pellet open porosity at BOL is nonexistent (Table 4.2.7.a-1).

4.2.7.2.a Effect of Extended Burnup on Burnable Poison Rod Behavior

Al_2O_3-B_4C Pellet Swelling. The swelling of Al_2O_3-B_4C pellets is strongly fluence dependent; therefore, the mechanical behavior of the burnable poison rod is affected by extended burnup. While the cladding may not be strained because of the large diametral gap in the new designs, the rod void volume will be decreased by the diametral and axial swelling of the pellets.

Gas Release. As discussed in Reference 1, helium is generated and released primarily in the first cycle of irradiation when the poison rod is operating at its highest temperature. Extended burnup, therefore, will not result in significant additional helium release. This behavior has already been verified by gas release measurements on burnable poison rods exposed for up to 4 cycles.

Axial Growth and Diametral Creep. Extended-burnup operation will result in additional elongation of the burnable poison rods. As discussed in Reference 1, the growth of the poison rods is no more limiting than the growth of the fuel rods.

The increment of diametral cladding creep associated with extended-burnup operation should be extremely small due to low cladding temperatures and low differential pressure across the cladding during this period of time. Full diametral contact between the pellets and cladding is not predicted so outward creep of the cladding due to swelling of the pellets is not expected.

Rod Internal Pressure. Internal pressure will increase during extended burnup operation due to a reduced void volume within the rod caused principally by pellet swelling. Rod growth and creepdown are second order effects on the void volume when compared to pellet swelling, but are accounted for. No additional gas is predicted to be released from the pellets due to extended burnup .

4.2.7.3.a <u>Evaluation of Burnable Poison Rod Behavior</u>

Well defined models exist for all fluence-dependent and time-dependent aspects of burnable poison rod behavior. When used in combination with the design improvements in the extended-burnup poison rod designs, they will demonstrate that there is margin to the strain, clearance, and internal pressure criteria for the poison rods.

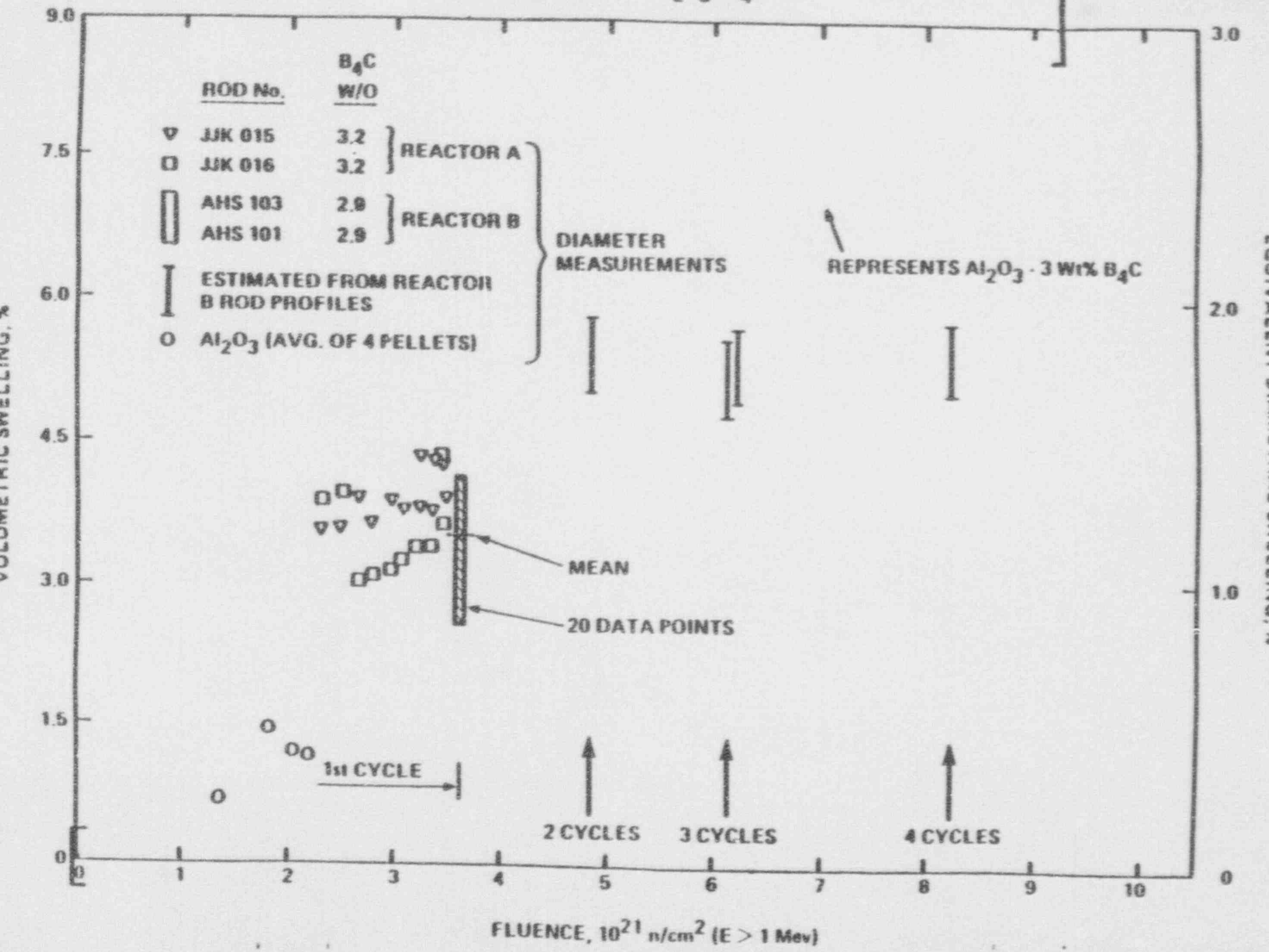
Table 4.2.7.a-1

Burnable Poison Rod Details

Parameter	Early 14x14 Design	Extended Burnup 14x14 Design	Early 16x16 Design	Extended Burnup 16x16 Design
Pellet O.D., in.	0.376-0.379	0.362	0.310	0.307
Cladding O.D., in.	0.440	0.440	0.382	0.382
Cladding I.D., In.	0.388	0.384	0.332	0.332

*Expressed as a percent of the total pellet volume.

FIGURE 4.2.7.a-1
SWELLING OF $Al_2O_3 - B_4C$



CONCLUSION

The objective of this report is to justify the validity of C-E methods and models concerning the 16x16 fuel assembly design and safety analysis for 1-pin burnups up to 60 MWD/kg. The present C-E licensing document on fuel burnup limits (Reference 1) justifies a 1-pin limit of 52 MWD/kg. The data presented in this report justify the extension of this 1-pin limit to the new 1-pin limit required by the implementation of longer fuel cycles, 60 MWD/kg. As such, the overall and individual conclusions presented in Reference 1 are shown to be valid for the extension of the 1-pin burnup limit to 60 MWD/kg for 16x16 fuel assembly designs.

The conclusions of this report regarding fuel assembly length change and shoulder gap change are applicable to Combustion Engineering 16x16 fuel assembly designs employing [ ].

Also, since the various fuel performance topics discussed in Reference 1 have no explicit dependence on batch average burnup, the batch average discharge limit of Reference 1 is no longer required and can be deleted.

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Appendix A

Response to NRC Questions on
Topical Report
"Verification of the Acceptability of a 1-Pin Burnup
Limit of 60 MWD/kgU for Combustion Engineering
16x16 PWR Fuel"

QUESTIONS and RESPONSES

Question #1:

The burst stain data from Fort Calhoun cladding with local burnup levels between 55 to 63 MWd/kgM, presented in Section 4.1.5.a of the topical report, show very low cladding strains between 0.03 to 0.1%. Section 4.2.II.A.2(g) of the Standard Review Plan (SRP) (Reference 1), which addresses "acceptance criteria" to preclude pellet/cladding interaction (PCI) failures, states that uniform strain (elastic plus plastic) of the cladding should not exceed 1% for normal operation and anticipated operational occurrences (AOOs). This strain limit has also traditionally been applied as a limit for cladding strain in Section 4.2.II.A.1(a) of the SRP. The cladding burst data from Fort Calhoun suggests that Combustion Engineering (C-E) fuel cladding may fail at uniform strains significantly below the 1% strain limit recommended in the SRP. Therefore, several questions arise from the data:

(a) How applicable are the burst tests and measured strains from the Fort Calhoun cladding to failure mechanisms from normal operation and AOOs in C-E commercial reactors? This response should address those failure mechanisms identified in Section 4.2 of the SRP. If this data is applicable to these failure mechanisms or if their applicability is unclear, please address the following additional questions.

(b) Should the uniform strain limit of C-E cladding for normal operation and AOOs be decreased to a level below 1% when local burnups exceed 55 MWd/kgM?

(c) Will the fuel cladding become even more embrittled at local burnups above 63 MWd/kgM?

(d) Will fuel failures become more frequent as the number of fuel rods that exceed local burnups of 55 MWd/kgM increases from commercial operation?

QUESTIONS and RESPONSES

Response to Question #1

Use of the high strain rate burst test data to evaluate the cladding strain capability against pellet/cladding interaction (PCI) failure caused by normal operation or AOO's is conservative for the following reason:

Strain rates anticipated during normal operation and AOO's are expected to be lower than those employed in the burst testing. At lower strain rates, the material ductility is expected to be higher.

In addition, it is to be noted that the data presented in Section 4.1.5.a (Page 21, first paragraph) of the topical report refer to uniform plastic circumferential burst strain of fuel cladding. These values do not contain the elastic component. At local burnup levels of 54.7 to 62.5 MWd/kgU, the measured uniform plastic strains ranged from 0.03 to 0.11%. The uniform plastic strains were calculated by subtracting the elastic component from the total uniform strain corresponding to the maximum pressure point on the pressure-volume expansion curve for the burst specimen (corrected for the p-Δv response of the system including the specimen). Elastic strain capability of the cladding needs to be added to the uniform plastic strain to obtain the total uniform strain (elastic plus plastic strain) capability of the material. Since the elastic strain of the sample cannot be accurately determined from the test data, an estimate of the sample elastic strain was obtained as follows: The measured yield strength of cladding in the same burnup range as quoted above ranged from 115 to 125 ksi. The reported Young's modulus of non-irradiated Zircaloy at 600°F is ~10×10^6 psi (Reference 2). Assuming no changes in the Young's modulus as a result of irradiation, the elastic strain capability of the cladding is estimated to be at least [ ]. Therefore, the above data show that the total (elastic plus plastic) strain capability of the cladding ranges from [ ]. Thus, the data on irradiated cladding show that the measured strains exceed the 1% minimum limit recommended in the SRP.

Response to Question #1 (continued)

Consideration of the following factors further reduces concern for PCI-related failures at extended burnups.

1. At extended burnups, the heat rates of the fuel rods are significantly lower than the heat rates earlier in life. The heat rates at extended burnups (>52 MWd/kgU) are expected to be significantly lower than the power threshold for PCI failures.

2. As a result of the establishment of tight fuel pellet cladding contact at high burnups, an interface interaction layer (main constituents Zr, U, and O) forms between the fuel pellet and Zircaloy cladding. This interface layer helps to distribute the mechanical load exerted by the fuel pellet on the cladding. Once the interface layer forms, the application of a concentrated localized stress (necessary for the PCI failures) is less likely. This layer also improves heat transfer between the fuel pellet and cladding. As a result, fuel-swelling induced stresses in the cladding are expected to be reduced in the case of power transients.

3. C-E has successfully irradiated fuel rods to local burnups of about 60 MWd/kgU in three commercial PWRs: ANO-2, Calvert Cliffs-1, and Fort Calhoun. The C-E fuel exhibited satisfactory performance both during the normal operation to 60 MWd/kgU burnup and subsequent post-irradiation handling. Failures associated with reduced ductility of the irradiated material were not observed. This C-E experience indicates that the probability of cladding failure due to reduced ductility is low and setting a strain limit below 1% is not required.

In summary, the burst test conditions are more severe for ductility considerations compared to the conditions that are analyzed for normal operation and AOO's. However, even under the more severe conditions

Response to Question #1 (continued)

imposed during the burst tests, the C-E cladding mechanical property data exhibited total uniform (elastic plus plastic) strains higher than the minimum 1% limit. Therefore, it is concluded that the failure strain of C-E cladding for every failure mechanism identified in Section 4.2 of the SRP will be greater than 1%.

References

1. U.S. Nuclear Regulatory Commission, July 1981. "Section 4.2, Fuel System Design." Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants--LWR Edition. NUREG-0800, Revision 2, U.S. Nuclear Regulatory Commission, Washington, D.C.

2. D. B. Scott, "Physical and Mechanical Properties of Zircaloy-2 and Zircaloy-4", WCAP-3269-41, May 1965.

QUESTIONS and RESPONSES

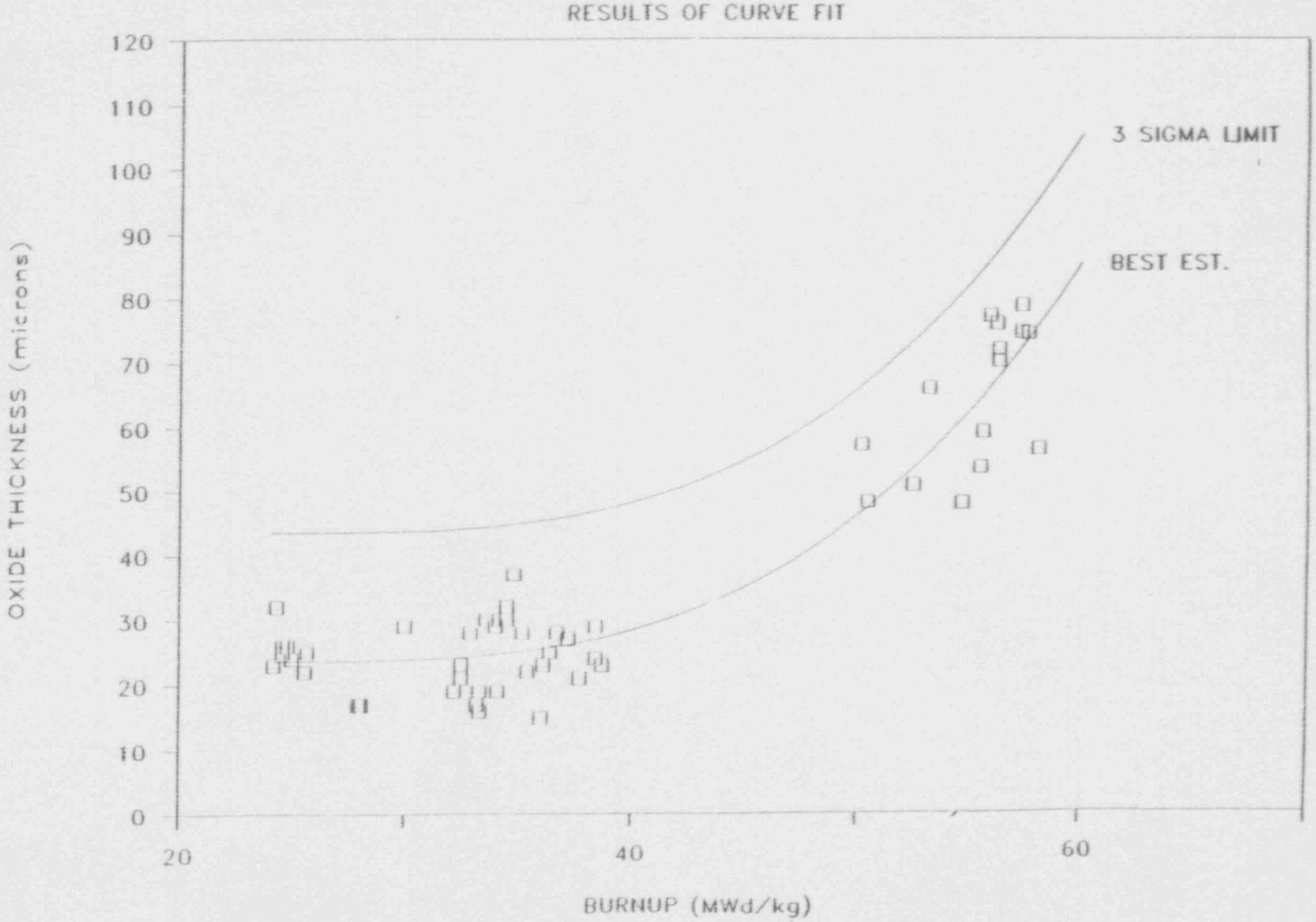
Question #2

Please discuss how the standard deviation, σ, is calculated for oxide thickness at rod average burnup levels of 60 MWd/kgM from the 14x14 and 16x16 fuel rod data in Section 4.1.2.1.a. Should this data be separated because there are inherent differences between the corrosion behavior of 14x14 and 16x16 fuel rods or should they be combined because the differences in corrosion are not uniquely design dependent? Also, it appears that the estimates of $\bar{x} + 3\sigma$, provided in this section, assume that σ is independent of burnup, while the corrosion data in Figures 4.1.2.a-1 and 4.1.2.a-2 suggests that σ becomes larger at higher burnups. What impact would a more variable and larger calculated σ have on the performance analyses of the 16x16 fuel rods (e.g., cladding stress) at 60 MWd/kgM? Please provide the effect as a percentage change from the condition of no cladding wastage due to corrosion.

Response to Question #2

The Calvert Cliffs and Fort Calhoun data are shown in Figure 4.1.2.a-1 in the report for information. They were not included in the curve fit. The curve fit and the standard deviation were determined from just the ANO-2 fuel rod data. This is shown in the attached Figure 1, which plots the data, the curve fit, and the 3σ limit. Figure 1 shows that the 3σ line is a conservative bound to the ANO-2 fuel rod data and that the standard deviation for these data is not burnup dependent. As mentioned above, Figure 4.1.2.a-1 includes data from reactors other than ANO-2. Consequently, the spread noted in that figure reflects the influence of a variety of plant specific factors and not just burnup alone. These factors include differences in the fuel rod operating history (e.g., heat flux), coolant temperature, and the temperature at the oxide/metal interface.

Fig. 1. ANO-2 Oxide Data



QUESTIONS and RESPONSES

Question #3

The results of the cladding collapse calculation in Section 4.1.4.a are based on an assumed finite "hot" axial gap length in the fuel column of modern C-E designs. In this analysis it is implied from post-irradiation examination (PIE) data that "hot" axial gaps greater than this assumed size have a low probability of existing in C-E's 16x16 design, but no probabilities are calculated based on this PIE data. What is the probability of the C-E 16x16 design having an axial gap of this assumed size or greater? The probability may be calculated using PIE data of "cold" axial gap sizes measured from modern fuel designs other than C-E's 16x16 design, but these designs should be comparable in fuel length, densification characteristics, and density. A correction between "hot" and measured "cold" gap size is permissible but assumptions made in this correction should be stated. The calculated probabilities may also take into account that today's fuel designs typically form smaller axial gaps in the fuel column than previous "older" fuel designs because of changes in fuel fabrication. For example, three separate populations of axial gap size can be identified according to the following fuel characteristics 1) older densifying fuel, 2) older nondensifying fuel with low fuel densities (i.e., <94% theoretical density), and 3) newer nondensifying fuel with higher fuel densities (i.e., >94% theoretical density); with fuel with the latter characteristics displaying the smallest axial gap sizes following irradiation.

Response to Question #3

As discussed in Section 4.1.4.a, C-E performs cladding collapse calculations using very conservative input assupmtions. The criterion for selecting the length of the hot axial gap in the fuel column is that the value must be at least as large as the maximum predicted hot axial gap, at 95% probability and 95% confidence level. Predicted hot axial gaps are based on adjusting cold measured axial gaps to hot operating conditions.

Response to Question #3 (continued)

Axial gap data were obtained as a result of a post-irradiation examination of fuel from the San Onofre Unit 2 Cycle 3 core. This fuel is typical of the current generation 16x16 (and 14x14) Combustion Engineering high density nondensifying fuel. Thirty axial gap measurements were obtained from 17 fuel rods. These cold axial gap measurements were analyzed and "hot" axial gaps were determined by accounting for axial thermal expansion. The largest cold gap measured was 0.9 inches. It was calculated that thermal expansion of the fuel column during reactor startup reduces the largest cold gap to 0.3 inches at normal operating conditions.

The calculation of axial thermal expansion was based on an evaluation which considered the effects of axial variation in linear heat rate, local pellet-clad gap conditions (which affect fuel temperatures), changes in these parameters with time and/or burnup, and the existence of more than one gap (where applicable) in a single fuel rod. It was assumed that the fuel column segment below an observed gap would thermally expand axially upward to reduce the gap size. Once the axial gap is closed, the fuel column above the gap would be pushed upward, closing successive gaps, if they existed.

Based on the assumption that the distribution of the data is normal, statistical analysis of the hot gap data has been performed. The finite hot axial gap length assumed in the cladding collapse calculation is well in excess of that expected at a 95% probability and 95% confidence level.

Reference 3-1

3-1 CEN-386-P, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60/MWd/kg for Combustion Engineering 16x16 PWR Fuel", Combustion Engineering, Inc., June 1989.

Question #4

From the small number of FATES3B predictions of low temperature fission gas release data provided in Table 4.1.6.a-3, it appears that this code may be underpredicting this data by a small amount (i.e., 1 to 2% release when rod average burnups exceed 54 MWd/kgM). What is the effect, if any, on fuel performance calculations at low temperatures if the FATES3B code predicts 1% release when 2.5% release is the actual amount released?

Response to Question #4

An empirical model to predict fission gas release was developed by C-E for use in FATES3 and was described in CEN-161(B)-P, (Reference 4-1). The model accounted for the effects of temperature, burnup, and grain size, and was calibrated against the data from UO_2 fuel available at that time. The FATES3 verification data were limited to burnups up to 48 MWd/kgU. Although the burnup of the verification data base for FATES3 extended up to 48 MWd/kgU, the fuel centerline temperatures associated with the highest burnup data were characteristically below 1250°F. The NRC completed the safety evaluation and approved the FATES3 model for C-E safety analysis but also imposed a restriction on the grain size used for fission gas release calculations (FATES3A Reference 4-2). Once additional high-burnup, high-temperature experimental data became available, C-E analyzed these data and found that modifications to the FATES3 fission gas release model were required to predict these high-burnup data on a best-estimate basis. Changes were incorporated into FATES3 (forming the FATES3B version, Reference 4-3) to increase the burnup dependence, temperature dependence and modify the kinetics of grain growth. The model was specifically tuned to predict high-burnup, high-temperature fission gas release at power levels and temperatures characteristic of fuel performance licensing calculations at high burnup. Consequently, the FATES3B predictions of gas release from high-burnup (>54 MWd/kgU), low-power test rods reported in Table 4.1.6.a-3 of Reference 4-4, are slightly underpredicted, but by less

Response to Question #4 (continued)

than 0.5% on average. Based on the formulation of the model of Reference 4-3, the high temperature release is not affected by this low temperature underprediction.

For licensing calculations the power history data are selected to give conservatively high fuel temperatures, high fission gas release, and high hot internal gas pressure. Therefore, because fuel performance licensing calculations are not performed at low temperature, the slight underprediction of the data in Table 4.1.6.a-3 (Reference 4-4) would have an insignificant effect on licensing calculations.

However, the impact of an additional 1.5% fission gas release at high burnup was evaluated. The degradation in gap conductance due to the additional gas release results in an insignificant increase in fuel temperatures, on the order of 3-5°F at 7-8 Kw/ft. The increase in rod internal pressure at 55.0 MWd/kgU due to an additional 1.5% fission gas release is less than 100 psi (75 psi for a typical low temperature Calvert Cliffs test rod). However, it should be noted that the average underprediction is only on the order of 0.5% as discussed above.

References:

4-1. CEN-161(B)-P, "Improvements to Fuel Evaluation Model", Combustion Engineering, Inc., July 1981.

4-2. CEN-161(B)-P-A, "Improvements to Fuel Evaluation Model", Combustion Engineering, Inc., August, 1989.

4-3. CEN-161(B)-P Supplement 1-P, "Improvements to Fuel Evaluation Model", Combustion Engineering, Inc., April 1986.

Response to Question #4 (continued)

4-4. CEN-386-P, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWd/kg for Combustion Engineering 16x16 PWR Fuel", Combustion Engineering, Inc., June, 1989.

QUESTIONS and RESPONSES

Question #5

In Section 4.2.1.a, on guide tube wear, it is stated that References 4.2.1.a-2 and 4.2.1.a-3 have verified the conservatisms in C-E analytical predictions of guide tube wear for these assemblies. Please provide a comparison of measured and predicted maximum wear for 16x16 unsleeved assemblies along with their burnup levels. What is the maximum wear predicted for the C-E 16x16 assemblies at the maximum residence times expected for the burnup levels requested?

Response to Question #5

Figure 4.2.1.a-1 (attached) provides a comparison of measured and predicted guide tube wear for unsleeved System 80 fuel assemblies (System 80 is the only unsleeved 16x16 fuel design). Assessments of the measured guide tube wear signals (voltage readings from Eddy Current Testing) employed conservative assumptions to maximize the calculated volume loss associated with the wear indications. The measured wear volumes shown in Figure 4.2.1.a-1 are the maximum volumes from any guide tube measured during the two inspection campaigns (eighty guide tubes inspected at Palo Verde 1 and forty guide tubes inspected at Palo Verde 2).

Inspection of Figure 4.2.1.a-1 shows significant margin between the maximum measured wear volumes and the corresponding predicted wear volumes. The maximum predicted wear volume at the maximum residence time associated with the extended burnup levels is [ ]. Analyses have been performed that demonstrate that the minimum amount of volume loss that a guide tube could sustain without violating any design criteria is [ ]. The unsleeved System 80 design is, therefore, concluded to be acceptable for operation to the extended burnup levels since the maximum predicted wear is less than the minimum wear necessary to violate any design criteria and since there is significant margin between the maximum measured wear volumes and their associated predictions.

FIGURE 4.2.1.a-1

COMPARISON OF ME[illegible]ED AND PREDICTED
GUIDE TUBE [illegible]R VOLUMES

WEAR VOLUME, CUBIC INCHES

OPERATING TIME, HRS

<u>Question #6</u>

Section 4.2.2.a addresses the shoulder gap between the top of the fuel rods and the bottom of the upper-and-fitting, but does not address the possibility of the assembly hold-down spring bottoming out due to assembly growth. Please demonstrate that the assembly hold-down spring does not bottom-out at the assembly burnup level requested. Also, what is the predicted gap margin for preventing bottoming-out of the hold-down spring?

<u>Response to Question #6</u>

Section 4.2.2.a addresses the evaluation of shoulder gap change and fuel assembly length change. Analytical models are presented along with post-irradiation data. Based on a comparison of the models to the data, it is concluded that the models can be used to conservatively predict the shoulder gap changes and fuel assembly length changes to extended burnups.

The intent of Section 4.2.2.a is to provide justification of the models used in evaluating the adequacy of 16x16 fuel assembly designs for shoulder gap change and fuel assembly length change. Margins were not presented because they vary for the different 16x16 fuel designs. Cycle specific evaluations are done for each plant to verify that, using these models, the specific fuel designs being loaded in the cycle are capable of operation without shoulder gap closure or bottoming out of the fuel assembly (i.e. maintaining holddown spring clearance to solid height and upper end fitting post to upper guide structure clearance). Based on the models presented in Section 4.2.2.a, limiting EOL clearances associated with fuel assembly length change for 16x16 fuel designs currently being fabricated are [typically at least 0.3 inches] at an assembly burnup consistent with a peak rod burnup of 60,000 MWD/MTU.

QUESTIONS and RESPONSES

Question #7

What is the maximum boron-carbide content, in weight percent, of the alumina-boron carbide pellets in the burnable poison rods for the 16x16 design? Have additional post-irradiation examinations been performed on burnable poison rods (such as for helium release, internal void volumes, pellet swelling, etc.) since those examinations presented in CENPD-269-P, Revision 1-P?

Response to Question #7

The maximum boron carbide content that has been used in alumina-boron carbide pellets in the 16x16 design is 3.76 weight percent. This level is slightly higher than the levels for which irradiation performance characteristics (helium release, internal void volume, and pellet swelling) were determined, as reported in CENPD-269-P, Revision 1-P. The C-E model accounts for the effect of higher levels of boron carbide by increasing the total volume of helium gas released and by increasing the pellet swelling. No additional post-irradiation examinations have been performed or are planned.