WCAP - 14204-A

WESTINGHOUSE FUEL CRITERIA EVALUATION PROCESS

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S. L. Davidson, Editor

Approved:

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SECTION A

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

July 27, 1994

Mr. N. J. Liparulo, Manager
Nuclear Safety and Regulatory Activities
Westinghouse Electric Corporation
P. O. Box 355
Pittsburgh, Pennsylvania 15230-0355

Dear Mr. Liparulo:

SUBJECT: ACCEPTANCE FOR REFERENCING OF TOPICAL REPORT WCAP-12488, "WESTINGHOUSE FUEL CRITERIA EVALUATION PROCESS" (TAC NO. M77257)

We have reviewed the subject topical report of April 1990, and your responses of January 8 and July 17, 1992; February 8 and August 25, 1993; and March 17 and May 20, 1994, to our requests for additional information. On the basis of our review, we conclude that WCAP-12488 is an acceptable basis for the fuel assembly design criteria evaluation process for licensing applications. Enclosed is our safety evaluation report (SER), which details the basis for and limitations of our approval. Our evaluation applies only to matters described in the topical report.

In accordance with procedures established in NUREG-0390, Westinghouse should publish accepted versions of this topical report, proprietary and non-proprietary, within 3 months of receipt of this letter. The accepted versions shall include an "A" (designating accepted) after the report identification symbol.

Should our acceptance criteria or regulations change so that our conclusions as to the acceptability of the report are no longer valid, applicants referencing this topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued applicability of the topical report without revision of their respective documentation.

Sincerely,

Martin J. Virgilio, Acting Director
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

Enclosure:
WCAP-12488-P Evaluation

SECTION B

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

ENCLOSURE

SAFETY EVALUATION OF WESTINGHOUSE ELECTRIC COMPANY

TOPICAL REPORT WCAP-12488

"WESTINGHOUSE FUEL CRITERIA EVALUATION PROCESS"

1.0 INTRODUCTION

By a letter from W. J. Johnson, Westinghouse Electric Corporation, to the U.S. Nuclear Regulatory Commission (NRC), dated April 2, 1990, Westinghouse submitted a topical report, WCAP-12489, "Westinghouse Fuel Criteria Evaluation Process," for NRC review.

In WCAP-12488, Westinghouse describes a process and criteria that it intends to apply to changes or improvements in existing fuel designs that will not require NRC review and prior approval when these criteria are satisfied. Westinghouse also will apply these criteria to adjustments or improvements of fuel performance design evaluation models, based on new data, without NRC review and approval. This approach to fuel design criteria is consistent with the staff position for other fuel vendors.

The NRC staff was supported in this review by its consultant, Pacific Northwest Laboratory (PNL). The staff has adopted the findings recommended in PNL's technical evaluation report (TER), which is attached, as modified by this safety evaluation report.

2.0 EVALUATION

The attached TER contains PNL's evaluation. During the course of preparing the SER, several issues arose such as gadolinia fuel, fretting wear, and DNBR. The three sections that follow provide additional comment on the respective issues.

2.1 Criteria for Revisions to Fuel Performance and Material Properties Models (TER Section 2.0)

The staff has previously approved the use of 6 w/o gadolinia fuel for Westinghouse. The staff also agreed with the Westinghouse proposed fuel mechanical design criteria to extend the application of the gadolinia fuel beyond 6 w/o as evaluated in the TER Section 2.0. In a letter from Northern States Power (NSP) Company to NRC dated October 14, 1994, NSP stated that it will use 8 w/o gadolinia fuel manufactured by Westinghouse in its Prairie Island pressurized water reactor plant. Because the design evaluation of the 8 w/o gadolinia fuel involves the Westinghouse fuel design criteria, the staff sent Westinghouse a request for additional information (RAI) on February 2, 1994, about the neutronic calculations and the implications of a licensee electing to use its own neutronic methodology to analyze Westinghouse gadolinia fuel.

In a letter dated March 17, 1994, Westinghouse proposed a process for extending the applicability of NRC approved Westinghouse neutronic methods for gadolinia fuel to concentrations greater than 6 w/o. Predicted model performance is to be compared to measured data from lead test assemblies and/or complete cores containing fuel with the higher gadolinia concentration, and evaluated by a statistical process to determine the analytical uncertainty of the model. Westinghouse also proposed that any licensee approved to use Westinghouse or its own design methods and codes for gadolinia neutronic analyses be permitted to extend the approval to higher gadolinia concentration for Westinghouse fuel by the same process. The staff reviewed these procedures and agreed with the Westinghouse approach. We concluded that the proposed application of approved neutronic methodology by Westinghouse or by licensees to evaluate Westinghouse fuel with higher gadolinia concentration is acceptable.

2.2 Fretting Wear (TER Section 3.4)

The NRC staff issued Information Notice 93-82 concerning recent fuel failures caused by several mechanisms. One of the failure mechanisms was grid-to-rod fretting wear phenomenon involved the use of VANTAGE 5H fuel and assembly locations adjacent to the core baffle. Westinghouse investigation revealed that under certain operating conditions the VANTAGE 5H fuel was subject to vibration that could damage the fuel rods. In light of this finding, Westinghouse submitted revised fretting wear design evaluation criteria in a letter dated August 25, 1993, from N. J. Liparulo (Westinghouse) to USNRC. The revised criteria included a commitment by Westinghouse to test a new fuel assembly for flow-induced vibration over a wide range of operating conditions. The staff reviewed the revised criteria and finds them acceptable for fretting wear evaluation of a new design.

2.3 Overheating of Cladding (TER Section 4.3)

During our review of the VANTAGE 5H fuel fretting issue, the staff noted that the original VANTAGE 5H fuel submittal, described in Addendum 2 to WCAP-10444-P-A, was incomplete with respect to the departure from nucleate boiling (DNB) testing of grid spacers. Specifically, the submittal indicated that the WRB-1 critical heat flux (CHF) correlation was appropriate for the VANTAGE 5H fuel, but the WRB-2 was appropriate if VANTAGE 5H with intermediate flow mixer grids (IFMs), characteristic of AP600 (advanced reactor) fuel, were used. However, confirmatory DNB testing was to be performed for the AP600 program. The staff questioned whether these confirmatory tests had been completed.

In a letter dated July 1, 1994 from N.J. Liparulo to USNRC (NTD-NRC-94-4186), Westinghouse stated that DNB testing has been performed to confirm the WRB-2 correlation applicability for the AP600 fuel design (with IFMs). In addition, Westinghouse indicated that final results also confirmed that WRB-1 remained applicable to VANTAGE 5H fuel (no IFMs). However, analysis of IFM data that included rotated IFM grids revealed that there was a significant difference in DNB performance between the aligned (non-rotated) and rotated IFM grids. Based on these data, Westinghouse, in another July 1, 1994 letter (NTD-NRC-94-4185), added pertinent mixing vane design characteristics to the list of geometric parameters that must be represented by the existing CHF data or new CHF tests would be required.

On the basis of our TER and the above evaluation, the staff concludes that the Westinghouse proposed criteria for confirming the applicability of CHF correlations to modified fuel designs are acceptable.

3.0 CONCLUSIONS

The staff has reviewed the Westinghouse fuel design criteria evaluation process described in WCAP-12488 and finds it acceptable for licensing applications up to 60,000 MWD/MTU rod average exposure.

TECHNICAL EVALUATION REPORT OF THE
TOPICAL REPORT WCAP-12488, "WESTINGHOUSE
FUEL CRITERIA EVALUATION PROCESS"

C. E. Beyer

March 1993

Prepared for
Reactor Systems Branch
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
under Contract DE-AC06-76RLO 1830
NRC FIN I2009

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Richland, Washington 99352

ACRONYMS

AOO	Anticipated Operational Occurrence
ASTM	American Society for Testing and Materials
CFR	Code of Federal Regulations
CHF	Critical Heat Flux
DNB	Departure from Nucleate Boiling
ECCS	Emergency Core Cooling System
FCEP	Fuel Criteria Evaluation Process
GDC	General Design Criterion
LOCA	Loss-of-Coolant Accident
LTA	Lead Test Assembly
NRC	U.S. Nuclear Regulatory Commission
PAD	Performance Analysis and Design
PCI	Pellet/Cladding Interaction
PCT	Peak Cladding Temperature
PNL	Pacific Northwest Laboratory
PWR	Pressurized Water Reactor
RIA	Reactivity Initiated Accident
SAFDL	Specified Acceptable Fuel Design Limit
SER	Safety Evaluation Report
SRP	Standard Review Plan
SSE	Safe-Shutdown Earthquake
TER	Technical Evaluation Report
W	Westinghouse Electric Corporation

CONTENTS

1.0 INTRODUCTION

In order to support customer needs and remain competitive, the fuel vendors are continually improving their fuel designs. Generally, the changes in design are made with approved methodologies. The regulatory procedures to qualify and approve the new designs are standard. However, the review and approval of these new designs place a burden on the staff resources.

Recently, the U.S. Nuclear Regulatory Commission (NRC) staff proposed that a set of acceptance criteria, to be satisfied by new fuel designs, be established for each fuel vendor. Once the acceptance criteria are approved, the fuel designs or changes satisfying the criteria would not require explicit staff review. Satisfaction of the acceptance criteria would be sufficient for approval by reference to the acceptance criteria. Also, the NRC staff requires that the acceptance criteria be entirely nonproprietary so that any interested party will have access to the acceptance criteria. The objective of this approach is to expedite the review process and reduce the staff and industry resources needed for review of new fuel designs.

In response to the NRC staff's proposal, the Westinghouse Electric Corporation (W) has submitted to the NRC a topical report, entitled "Westinghouse Fuel Criteria Evaluation Process" (WCAP-12488), for review and approval (Reference 1). This report describes the process and criteria that W intends to apply to changes in existing fuel designs that will not require NRC review and approval as long as these criteria are satisfied. W intends to apply these criteria to all of their current fuel designs. The W analysis methodologies that are used for determining that the specific fuel design criteria are satisfied have also been identified and previously approved by the NRC and, therefore, have not been reviewed in detail during this review, but will be discussed briefly. Also provided in Section 7.0 of WCAP-12488 (a new section added to the report during the review process) are those criteria for making adjustments to fuel performance and material property models based on new data without NRC review and approval. The original submittal of WCAP-12488 also requested an extension in the fuel burnup limit for W fuel designs, however, this request was withdrawn by W and will be addressed in the review of WCAP-12610. Topical report WCAP-12610 has been reviewed and approved up to a rod-average burnup of 60 GWd/MTM; however, the review for burnup levels beyond this current level are pending. Therefore, the issue of an extension in fuel burnup limit for W fuel designs has not been reviewed nor approved as a part of this submittal. Therefore, this review and Technical Evaluation Report (TER) is only applicable to currently approved burnup level of 60 GWd/MTM (rod-average).

Pacific Northwest Laboratory (PNL) has acted as a consultant to the NRC in this review. As a result of the NRC staff and their PNL consultant's review of this topical report, a list of questions were sent by the NRC to W requesting clarification of specific design criteria and licensing analyses (Reference 2). W has provided responses to these questions in Reference 3. As noted above, W has submitted (Reference 4) a Section 7.0, entitled "Fuel Performance and Material Properties Model," for inclusion in WCAP-12488 that was not included in the original submittal of WCAP-12488. Following further discussions, W submitted a revision to one subset of the questions and a revised Section 7.0 (Reference 5).

This review of WCAP-12488 was based on those licensing requirements identified in Section 4.2 of the Standard Review Plan (SRP) (Reference 6). The objectives of this review of fuel design criteria, 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 fuel system damage is never so severe as to prevent control rod insertion when it is required, 3) the number of fuel rod failures is not underestimated for postulated accidents, and 4) the coolability is always maintained. A "not damaged" fuel system is defined as fuel rods that do not fail, fuel system dimensions that remain within operational tolerances, and functional capabilities that are not reduced below those assumed in the safety analyses. Objective 1, above, is consistent with General Design Criterion (GDC) 10 [10 Code of Federal Regulations (CFR) 50, Appendix A] (Reference 7), and the design limits that accomplish this are called specified acceptable fuel design limits (SAFDLs). "Fuel rod failure" means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached. Fuel rod failures must be accounted for in the dose analysis required by 10 CFR 100 (Reference 8) for postulated accidents. "Coolability," which is sometimes termed "coolable geometry," means, in general, that the fuel assembly retains its rod-bundle geometrical configuration with adequate coolant channels to permit removal of residual heat even after a severe accident. The general requirements to maintain control rod insertability and core coolability 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, Section 50.46.

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 are applied to postulated accidents. Specific fuel damage or failure mechanisms are identified under each of these categories in Section 4.2 of the SRP. The W fuel criteria evaluation process (FCEP) is discussed in this report under each fuel damage of failure mechanism listed in the SRP. A brief discussion is also provided of the previously approved analysis methods that are used to assure that the design limits and, thus, SAFDLs are met for a particular design application. W has also proposed that they be able to make minor adjustments to some analysis methods or codes without NRC approval. A discussion will be provided in Section 2.0 on these criteria prior to the discussion of design limits.

The purpose of the design criteria or limits are to provide limiting values that prevent fuel damage or failure with respect to each mechanism. Reviewed in this report is the applicability of the W FCEP criteria/limits to the W fuel designs up to currently approved burnup levels. These approved design criteria/limits, along with certain definitions for fuel failure, constitute the SAFDLs required by GDC 10.

W was questioned (Reference 2) on when a fuel design or analytical model change will or will not need to be submitted for review before implementation in a core reload. A summary of W's response (References 3 and 5) is that an NRC review will not be required if a design, material, or analytical model change meets all of the following criteria:

1. The change in the integrated fuel performance code response for: a) maximum fuel average temperature, b) end of life rod internal pressure, and c) cladding strain, relative to that predicted using the NRC approved PAD 3.4 code version for established benchmark cases will not exceed the limits defined in Table 7.1.

2. The change in the predictive capability for each model which is modified, relative to the reference performance model data base for the NRC approved PAD code version, will be within the value specified for each model in Table 7.2. Only those models specified in Table 7.2 will be subject to revision under the Fuel Criteria Evaluation Process.

3. New fuel performance or material property data sets to be used in developing a modified performance or material property model will be evaluated relative to the current NRC approved model prior to incorporation into an updated model development data base. Only those data sets which meet the limits specified in Table 7.2 relative to the current approved model will be used in model adjustments under the Fuel Criteria Evaluation Process.

4. New fuel performance models will not be introduced using the Fuel Criteria Evaluation Process. Any new performance phenomena to be modeled in a revised PAD version will be submitted to the NRC for explicit review and approval. Subsequent to NRC approval of a new model, future revision to that model will be subject to the Fuel Criteria Evaluation Process.

5. The Fuel Criteria Evaluation Process will not be used to extend the range of applicability of any fuel performance or material property model to burnup levels greater than a proprietary value beyond the target lead rod average licensed burnup limits approved for W fuel by the NRC.

6. The Fuel Criteria Evaluation Process will not be used to extend the applicability of fuel performance of material property models to new materials, where new materials are defined as materials which have not explicitly been reviewed and approved for use in reactors by the NRC.

Therefore, an NRC review will be required if a design, material or analytical model change fails to meet any one of the above criteria. PNL concludes that these criteria limit changes to W fuel designs and analytical models as intended and, therefore, are acceptable. It should be noted that W has agreed to submit to NRC for information their design and model changes that meet the criteria, as defined in WCAP-12488, even though they will not be subjected to NRC review.

Those criteria that define how an analytical model can be changed without NRC review and approval, as proposed in Section 7.0 of Reference 5 are addressed in Section 2.0. The fuel damage and failure mechanisms are the same as those in Section 4.0 of the SRP and are addressed in Sections 3.0 and 4.0, respectively, while fuel coolability is addressed in Section 5.0 of this report.

2.0 CRITERIA FOR REVISIONS TO FUEL PERFORMANCE AND MATERIAL PROPERTIES MODELS

W has submitted (Reference 4) a Section 7.0 to WCAP-12488 that was not included in the original submittal of this topical report (Reference 1). W has subsequently revised Section 7.0 with the submittal of Reference 5. The intent of Section 7.0 is to allow adjustments to specific W fuel performance and material property models as new data for these models are collected. If the new data collected identify new mechanisms or significant changes in fuel performance, it is the intent of NRC and W that this new information will be submitted to NRC for review and approval. Therefore, the revised Section 7.0 presented in Reference 5 identifies those criteria that limit revisions to fuel performance models in the PAD 3.4 code and material property models without NRC review and approval. These six W criteria or tests of significance are defined in the Introduction (Section 1.0) of this TER and will be briefly discussed in this section.

The first W criterion limits the change in the integrated response of the PAD 3.4 fuel performance code. The change in the integrated response of the PAD 3.4 fuel performance code will be limited by W for the following predictions: 1) maximum fuel average temperature, 2) end-of-life rod internal pressure, and 3) cladding strain. The specific limits are identified in Table 7.1 of Reference 5 and the benchmark cases to which these specific limits are referenced are defined in Section 7.0. All W changes must be included for the tests on integrated PAD 3.4 changes relative to the NRC approved PAD code version rather than a test on integrated performance based on individual model revisions. These W limits on the integrated performance predictions of the PAD fuel performance code are found to limit changes to the code within the bounds of acceptable uncertainties of the code.

The second W criterion limits the change in the individual revised W models such that they will be tested by W to determine that the revised model adequately fits the previously approved data base of the current NRC approved model. A comparison of the predictions of the revised model to the reference data base of the current NRC approved model will be within the value specified for this model in Table 7.2 of Reference 5. PNL concludes that these values limit changes to W models as intended and, therefore, are acceptable.

The third W criterion also limits changes in individual fuel performance or material properties models by testing to determine if the new data to be used in a model revision are significantly different from the NRC approved data base and model as specified by the limits in Table 7.2. The intent of this test is to identify new physical mechanisms that are not included in the current model but are apparent in the new data base. If new mechanisms are identified, the model needs to be reformulated and submitted for review. The new data set will be compared to the NRC approved model and must meet the limits specified in Table 7.2. PNL concludes that these limits do limit changes to W models as intended and, therefore, are acceptable.

The fourth criterion is that new fuel performance models will continue to be submitted to NRC for review and approval. PNL concludes that this criterion is consistent with NRC's intent to limit changes to W models and, therefore, is acceptable.

The fifth criterion is that W fuel performance and material properties models cannot be applied beyond a proprietary burnup limit without NRC review and approval. PNL concludes that this criterion is consistent with NRC's intent to limit changes to W models and, therefore, is acceptable.

The sixth criterion is that current W fuel performance and material properties models, approved by NRC for current or previous materials, cannot be applied to new materials without NRC review and approval. PNL concludes that this criterion is consistent with NRC's intent to limit changes to W models and, therefore, is acceptable.

It should also be noted that W will continue to apply fuel performance predictions from revised models in a conservative manner relative to W design criteria such that model uncertainties will conservatively bound 95% of the model data base for these analyses.

The criteria defined in Section 7.0 of Reference 5 that allow revisions to fuel performance and material properties models have been reviewed and PNL concludes they are acceptable based on the NRC intent that limited revisions can be implemented without NRC review and prior approval. It should be noted that W has agreed to submit these model revisions including revision to the PAD 3.4 fuel performance code, permitted in Section 7.0, to NRC for their information even though they will not be subjected to NRC review.

3.0 FUEL SYSTEM DAMAGE

The design criteria presented in this section should not be exceeded during normal operation and AOOs. The evaluation portion of each damage mechanism evaluates the analysis methods and analyses used by W to demonstrate that the design criteria are not exceeded during normal operation including AOOs for W designs.

3.1 DESIGN STRESS

Bases/Criteria - The W design basis for fuel assembly, fuel rod, burnable poison rod, and upper end fitting spring stresses is that the fuel system will be functional and will not be damaged due to excessive stresses. The design limit for fuel rod cladding stress under normal operation and AOOs of operation is that the volume averaged effective stress calculated with the Von Mises equation, considering interference due to uniform cylindrical pellet-to-cladding contact (caused by pellet thermal expansion and swelling, uniform cladding creep, and fuel rod/coolant system pressure differences), is less than the Zircaloy-4 and ZIRLO 0.2 percent offset yield stress with consideration of temperature and irradiation effects as described in References 9, 10, and 11. These reports have been approved by the NRC. PNL concludes that these criteria are applicable for Zircaloy-4 and ZIRLO cladding up to currently approved burnup levels for application to fuel design changes.

W has proposed that they be allowed to make small changes to their Zircaloy-4 and ZIRLO cladding yield strength relationships without NRC review and approval (Reference 5). The review of this proposal is provided in Section 2.0 of this report.

Evaluation - The Performance Analysis and Design (PAD) code, Version 3.4 (Reference 12), is used by W along with the Von Mises equation to assure that the above criteria is met. This code has been verified against fuel rod data with rod-average burnup levels up to approximately 62 GWd/MTM. The burnup level of 62 GWd/MTM at which this code has been verified should not be confused with the current burnup limit of 60 GWd/MTM (rod-average) at which W fuel designs have been approved. This code takes into account those parameters important for determining cladding stresses at extended burnups, such as pellet thermal expansion and swelling, cladding creep, and fuel rod/coolant system pressure differences. It is noted that W reduces their cladding thickness by a proprietary amount in their mechanical analyses (including cladding stress) to account for fabrication flaws and waterside corrosion as described in Reference 13. The NRC has approved the use of this code for licensing applications involving both Zircaloy-4 and ZIRLO clad fuel. PNL concludes that this code is acceptable for determining cladding stress on W fuel rod design changes up to their currently approved burnup limit.

3.2 CLADDING DESIGN STRAIN

Bases/Criteria - The W design basis for fuel rod cladding strain is that the fuel system will not be damaged due to excessive cladding strain. In order to meet this design basis, the W design limit for cladding strain during steady-state operation is that the total plastic tensile creep and uniform cylindrical fuel pellet expansion due to fuel swelling and thermal expansion is less than 1 percent from the unirradiated condition. For AOO transients, the design limit for cladding strain is that the total tensile strain due to uniform cylindrical pellet thermal expansion during the transient is less than 1 percent of the pretransient value. These design strain bases and limits are intended to preclude excessive cladding deformation during normal operation and AOOs. These limits are the same as used in Section 4.2 of the SRP and have been previously approved by the NRC (References 10 and 13) for application to W fuel designs up to current burnup limits.

From the above, PNL concludes that the strain limits proposed by W for the FCEP are acceptable. It should be noted, however, that the material property that could have a significant impact on the cladding strain limit at burnup levels beyond those currently approved is cladding ductility. The strain criterion could be impacted if cladding ductility were decreased, as a result of extended burnup operation, to a level that would allow cladding failure without the normal operation and AOOs cladding strain criteria being exceeded in the W analyses. This issue will be addressed in the review of WCAP-12610, where W has requested an extension in their fuel burnup limit.

Evaluation - The NRC approved W fuel performance code, PAD 3.4 (Reference 12), is used to assure that W fuel meets the above criteria. As noted in the Design Stress section, this code has been verified against fuel rod data with rod-average burnup up to approximately 62 GWd/MTM and takes into account those parameters important for determining cladding stresses and strains at extended burnup limits. PNL concludes that, this code is acceptable for determining cladding strains on W fuel rod designs up to their current burnup limit.

3.3 STRAIN FATIGUE

Bases/Criteria - The W design basis for fuel rod cladding fatigue is that the fuel system will not be damaged due to cladding strain fatigue. In order to assure that this design basis is met, W imposes a design limit for strain fatigue such that the fatigue life usage factor is less than 1.0. That is, for a given strain range, the number of strain fatigue cycles are less than those required for failure when a minimum safety factor of 2 on the stress amplitude or a minimum safety factor of 20 on the number of cycles, whichever is the more conservative, is imposed. This criteria is essentially the same as that described in Section 4.2 of the SRP and, thus, has been approved for application to all W fuel oesigns up to currently approved burnup levels.

As noted in the Cladding Design Strain section, the material property that could have a significant effect on cladding strain and, thus, strain fatigue at extended burnup levels is cladding ductility. However, as discussed above, extended burnup operation above the level currently approved will be addressed in the review of WCAP-12610.

PNL concludes that these design criteria are acceptable for application to W fuel design changes.

Evaluation - The NRC approved W fuel performance code, PAD 3.4 (Reference 12), is used to determine the strain range for the fatigue usage analysis. The Langer-O'Donnell fatigue model (Reference 14), with the empirical factors in the model modified in order to conservatively bound the W Zircaloy-4 data, is used with the strains from PAD 3.4 to assure that the above criterion is met. A description of this methodology and the W data base is presented in WCAP-9500 (Reference 15), which has been approved by the NRC. This strain fatigue methodology has also been found to be acceptable by NRC for ZIRLO clad fuel (Reference 10). The W methodology takes into account daily load follow operation and the additional fatigue load cycles that may result from extended burnup operation. In addition, the PAD 3.4 code accounts for those parameters important for determining cladding strains at extended burnup levels, see Sections 3.1 and 3.2 of this report. Therefore, PNL concludes that the methodology adequately models operational and material behavior parameters important for determining strain fatigue for W fuel designs up to currently approved burnup levels.

3.4 FRETTING WEAR

Bases/Criteria - Fretting wear is a concern for the fuel rod cladding. Fretting, or wear, may occur on the fuel rod cladding surfaces in contact with the spacer grids if there is a reduction in grid spacing loads in combination with small amplitude, flow induced, vibratory forces.

Although 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 W design basis for fuel rod fretting wear is that fuel rods shall be designed not to fail due to fretting wear during normal operation and AOOs events. In order to meet this basis, W uses a general guide for wall thickness reduction which is a percent of the original wall thickness (the specific value is proprietary) for evaluating cladding imperfections, including wear marks. W indicates that the cladding stress and fatigue limits, discussed here in Sections 3.1 and 3.3, apply to fretting wear. W has also indicated (Reference 13) in the past that fretting wear will not have a significant effect on cladding stresses and, thus, need not be considered in stress related analyses. As long as the fretting wear in W fuel designs is demonstrated to be below the W guideline for cladding imperfections as stated in Reference 13, fretting wear is considered to be acceptable. Therefore, PNL concludes that these design bases and criteria are acceptable for W fuel design changes up to their currently approved burnup limits.

Evaluation - W has stated that to assure satisfactory fretting wear performance of W fuel designs, they will use three methods of evaluation: 1) experimental data from in-reactor performance; 2) 1000 hour out-of-reactor wear testing; and 3) the use of previously established and NRC reviewed and approved fretting wear models. Previously established fretting wear models will only be used if fuel design changes do not change hydraulic or grid support conditions from those employed on previous designs. If the design changes impact the hydraulic conditions or rod support conditions and are outside of current experience, then the remaining two fretting wear testing methods are employed by W.

This W methodology for evaluating rod fretting wear has been used by W for a number of fuel design changes for a number of years. Since the adoption of this methodology, W has not experienced fretting wear failures due to fuel assembly design changes. Currently, fretting failures are principally due to errors in fabrication, grid damage due to fuel handling, or debris in the primary system. Therefore, PNL concludes that the W methodologies for evaluating fretting wear are acceptable.

3.5 OXIDATION AND CRUD BUILDUP

Bases/Criteria - The W design basis for cladding oxidation is that the fuel system will not be damaged due to excessive cladding oxidation. In order to preclude a condition of accelerated oxidation, W imposes specific temperature limits on the cladding. The temperature limits applied to cladding oxidation are that calculated cladding temperatures (at the oxide-to-metal interface) shall be less than a specific (proprietary) value during steady-state operation and AOOs transients (a higher limit is applied for AOOs transients). These criteria have been approved by NRC (References 10, 13, and 15) for previous W fuel designs up to current burnup levels; however, Section 4.2 of the SRP states that the effects of cladding oxidation and crud need to be included in safety and design analyses, such as for thermal and mechanical analyses. With this provision, PNL concludes that the W design criteria for oxidation and crud buildup are acceptable.

Evaluation - Section 4.2 of the SRP states that the effects of cladding crud and oxidation need to be addressed in safety and design analyses, such as in the thermal and mechanical analyses. The amount of cladding oxidation is dependent on fuel rod power, water chemistry control, and primary inlet coolant temperature, but the amount of oxidation and crud buildup increases with burnup and cannot be eliminated. Therefore, the extended burnup levels of todays fuel designs result in thicker oxide layers that provide an extra thermal barrier and cladding thinning that can affect the mechanical analysis. The degree of this effect is dependent on reactor coolant temperatures and the ability of the water chemistry program to control oxidation.

W currently has NRC approved oxidation models for both Zircaloy-4 and ZIRLO cladding (References 12 and 10, respectively), that are used for thermal analyses. In addition, W is required to use these models to verify that the amount of cladding oxidation for each fuel reload application is consistent with the cladding wastage amount assumed for mechanical analyses. PNL concludes that these models remain acceptable for fuel design applications that utilize these two cladding materials and where the reactor coolant temperatures and coolant chemistry of the oxidation data bound future applications.

W has proposed (Reference 5) that they be allowed to make small changes to both the Zircaloy-4 and ZIRLO cladding oxidation models without NRC review and approval. The review of this proposal is provided in Section 2.0 of this report.

3.6 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 (References 10, 13, 16, 17, and 18) this for W fuel designs up to current burnup levels. PNL concludes that this basis remains acceptable for W fuel designs up to current burnup levels.

Evaluation - The W methodology for evaluating fuel and burnable poison rod bowing in 14x14, 15x15, and 17x17 assembly designs has been addressed in approved References 10, 16, 17, 18 and 19. The W methodology for the analysis of rod bowing has been found to be conservative up to current burnup levels (Reference 13). Rod bowing has been found to be dependent on the distance between grid spacers and the rod moment of inertia. Therefore, PNL concludes that this analysis methodology is acceptable for W fuel designs as long as a fuel design change does not change the distance between grid spacers or the rod moment of inertia from those in previously approved W fuel designs.

3.7 AXIAL GROWTH

Bases/Criteria - Failure to adequately design for axial growth of the fuel rods can lead to fuel rod-to-nozzle gap closure and fuel rod bowing and possible failure. Failure to adequately design for assembly growth can lead to collapse of the assembly hold-down springs. The W design basis is that the fuel rods will be designed with adequate clearance between the fuel rod ends and the top and bottom nozzles to accommodate the differences in the growth of the fuel rods and the growth of the fuel assembly.

The W design limit for fuel rod growth, References 13 and 15, is that no interference between the fuel rods and the fuel assembly top and bottom nozzles will occur. These bases and design limits have been accepted by the NRC for current W fuel designs and PNL concludes that they are acceptable for W design changes.

Evaluation - W currently uses the same axial rod growth model for both Zircaloy-4 and ZIRLO clad rods, which has been approved for current fuel designs up to current burnup levels (Reference 10). In addition, the fuel assembly growth model for annealed Zircaloy thimble tubes and associated methodology have also been approved for current fuel designs (References 10, 12, and 13). PNL concludes that these fuel rod and assembly growth models are acceptable for W fuel design changes as long as the materials used for cladding and thimble tubes are not altered.

3.8 ROD INTERNAL PRESSURE

Bases/Criteria - The W design basis for fuel rod internal pressure is that the fuel system will not be damaged due to excessive fuel rod internal pressure. The W design limits used to meet this design basis are that the internal pressure of the lead rod in the reactor will be limited to a value below which could result in 1) the diametral gap to increase due to outward cladding creep during steady-state operation, and 2) extensive departure from nucleate boiling (DNB) propagation to occur. This design basis and the associated limits have previously been found acceptable by the NRC for current fuel designs and current burnup levels (References 13 and 20). PNL concludes they are also acceptable for design changes up to current burnup levels.

Evaluation - The models and methods used by W to evaluate whether their designs meet the above basis and limits are examined in this section. The models used by W are contained in the PAD 3.4 code (Reference 12), which has been approved by the NRC up to current burnup levels. The NRC review of this code paid particular attention to those parameters important to internal rod pressure predictions, i.e., the thermal and fission gas release models. PNL concludes that the PAD 3.4 code is acceptable for use in the evaluation of rod internal pressures when fuel rod design changes are instituted.

W has proposed (Reference 5) that they be able to make small changes to a number of models without NRC review and approval that are important to the evaluation of fuel rod pressures. Those models that are important to evaluating rod pressures and that fit under this proposal are the cladding creep, fuel densification/swelling, and fission gas release models. The cladding creep and fuel densification/swelling models are used to determine the maximum gas pressure limit in order to prevent the diametral gap from increasing due to outward cladding creep during steady-state operation. A concern was expressed to W that if the cladding creep model were decreased by a small amount and the fuel swelling model increased by a small amount, i.e., for the nonconservative direction, the possible net effect on raising the maximum rod pressure limit may be large. W responded that they performed a calculation taking maximum possible changes allowed for cladding creep and fuel swelling rod and the resulting change in the pressure limit was small, particularly when compared to the impact that fission gas release model changes have on rod pressures. Consequently, PNL concludes that the limits on changes to cladding creep and fuel swelling models in Section 7.0 will effectively limit the change in the maximum rod pressure limit. A further review of the W proposal to make small changes to these models without NRC review and approval is discussed further in Section 2.0 of this report.

3.9 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 W design criteria is that they meet this criterion for all normal operation and AOOs events with the exception of the turbine overspeed transient associated with a loss of external load. The NRC has accepted this condition in the past for current fuel designs as long as the affected fuel assemblies can be shown to reseat properly in the core plate without damage or other adverse effects during the turbine overspeed event (References 10, 13, 15, and 16). PNL concludes that this remains acceptable for future design changes as long as the same criteria are met.

Evaluation - The fuel assembly liftoff forces are a function of primary coolant flow, spring forces, and assembly dimensional changes. W calculates these liftoff forces based on pressure loss coefficients that are determined from hydraulic testing (References 15 and 16), best estimate flows, and core bypass flow taking into account the associated uncertainties from each of these hydraulic conditions. PNL concludes that this analysis methodology is also acceptable for future fuel design changes.

4.0 FUEL ROD FAILURE

In the following paragraphs, fuel rod failure thresholds and analysis methods for the failure mechanisms listed in the SRP will be reviewed. When the failure thresholds are applied for normal operation including AOOs, they are used as limits (and hence SAFDLs) because fuel failure under those conditions should not occur according to the traditional conservative interpretation of the GDC 10. When these thresholds are used for postulated accidents, fuel failures are permitted, but they must be accounted for in the dose assessments required by 10 CFR 100. The basis or reason for establishing these failure thresholds is thus established by GDC 10 and Part 100 and only the threshold values and the analysis methods used to assure that they are met are reviewed below.

4.1 HYDRIDING

Bases/Criteria - Internal hydriding as a cladding failure mechanism is precluded by controlling the level of hydrogen impurities in the fuel during fabrication; this is an early-in-life failure mechanism. The moisture level in the uranium dioxide fuel is limited by W to less than or equal to 20 ppm, and this specification is compatible with the American Society for Testing and Materials (ASTM) specification (Reference 21), 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 the NRC (References 10 and 13) PNL concludes that this limit is acceptable for future W fuel design changes.

W also has a design limit on the hydrogen pickup level of 600 ppm due to waterside corrosion up to the current burnup levels of 60 GWd/MTM rod-average. Cladding hydrogen pickup limits are required to prevent excessive degradation of cladding mechanical properties due to hydrogen embrittlement by the formation of zirconium hydride platelets when hydrogen is released during the cladding oxidation process. W has previously indicated (Reference 10) that their test results show that ZIRLO and Zircaloy-4 have essentially the same relationship of hydrogen pickup at an equivalent level of oxidation. In addition, W in-reactor and ex-reactor corrosion data have demonstrated that ZIRLO oxidizes at a slower rate than Zircaloy-4. W has also stated that process controls and texture acceptance tests assure that W cladding maintains the proper hydride orientation (Reference 10).

External hydriding due to waterside corrosion is a possible reason for the observed ductility decrease in Zircaloy-4 cladding at local burnup levels >55 GWd/MTM (References 22 and 23). Garde (Reference 24) has recently proposed that the ductility decrease is due to a combination of hydride formation and irradiation damage at these high burnup levels. The hydride levels observed in these references were below the W 600 ppm limit. PNL concludes that the W limit on cladding hydrogen pickup is acceptable for future W design changes up to the current burnup limit, but this will be a concern in the review of future burnup extensions.

Evaluation - External hydrogen uptake of Zircaloy-4 during normal reactor operation up to an extended burnup level of 60 GWd/MTM (rod-average) has been found to be satisfactory for current W designs (References 10, 13, 16, 17, and 18). The exception to this is when an abnormal amount of cladding oxidation is encountered that results in cladding failure. It is expected that ZIRLO cladding will have lower hydriding than Zircaloy-4 because ZIRLO is expected to have less oxidation, but W needs to continue to obtain more data up to a rod-average burnup level of 60 GWd/MTM and above in a commercial pressurized-water reactor (PWR) operating environment to confirm this, as noted in the NRC Safety Evaluation Report (SER) of Reference 10. PNL concludes that external hydrogen pickup for future design changes is acceptable as long as the cladding material does not change from those of Zircaloy-4 and ZIRLO and primary coolant temperatures are not increased above those currently employed in W plants.

4.2 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 W design basis that fuel rod failures due to flattening will not occur. In order to meet this design basis, W imposes a design limit for fuel rod cladding flattening such that the core residence time shall not exceed the calculated core residence time corresponding to a flattened rod frequency of 1.0. This design basis and its associated criterion are essentially the same as those specified in the SRP and have been approved for current W fuel designs (References 10, 13, 16, 17, 18 and 25). PNL concludes that this design basis and its associated criterion are acceptable for future design changes up to a rod-average burnup level of 60 GWd/MTM.

Evaluation - Extensive postirradiation evaluations by W have not shown any evidence of cladding collapse or large local ovalities in their current generation fuel designs. This is primarily the result of their use of prepressurized rods and stable fuel in current generation designs.

W utilizes an NRC approved Zircaloy-4 cladding collapse model (Reference 25) for both Zircaloy-4 and ZIRLO clad fuel to demonstrate that the in-reactor residence times will not result in the cladding collapse of current fuel designs. This method is very conservative in relation to the stable fuel designs employed in current designs because it assumes a gap has formed in the fuel column and allows the cladding to creep into this gap. This method is also conservative with respect to ZIRLO cladding because ZIRLO has a lower creep rate than Zircaloy-4. This method was approved for use with current W fuel designs (References 13, 16, 17, and 18), and fuel designs with ZIRLO cladding (Reference 10). PNL concludes that this W analysis model for cladding collapse is also acceptable for application to future fuel design changes.

4.3 OVERHEATING OF CLADDING

Bases/Criteria - The W design basis 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 DNB will not occur on a fuel rod having the minimum DNB ratio during normal operation and AOOs. This design basis is consistent with the thermal margin criterion of Section 4.2 of the SRP and, therefore, PNL concludes that it is acceptable for application to future W fuel design changes up to a rod-average burnup level of 60 GWd/MTM. Based on this design basis, W has defined DNB ratio limits for different DNB correlations and design configurations that have been reviewed and approved by the NRC and these are listed in Reference 1.

Evaluation - W utilizes several NRC approved codes and analysis methods to calculate the minimum DNB ratio values in the hot channels (approved references and listed in Reference 1) to provide assurance that there is at least a 95% at a 95% confidence level that the minimum DNB ratio will be greater than or equal to the correlation limit. The DNB correlation for an approved fuel design will be evaluated with respect to design changes that deviate from this approved W design. An existing NRC approved DNB correlation is considered to be valid by W for application to a new design when the new fuel assembly geometry is similar to or bracketed by the fuel assembly geometric parameters and correlation parameters of the critical heat flux (CHF) test data used to develop the approved DNB correlation. The relevant geometric parameters of the test data that are compared to a new assembly geometry are: rod diameter, thimble tube diameter, rod pitch, heated length, gridded to ungridded cell flow area, grid spacing, and mixing vane design. Typical correlation parameters of the test data that are compared to those of a new assembly are pressure, local mass velocity, axial power shape, local quality, heated length (inlet to CHF location), grid spacing, equivalent diameter, equivalent heated hydraulic diameter, and distance from the last grid to the CHF location. Exact correlation parameters are a function of the DNB correlation in question. Specific mixing vane designs encompass the ratio of vane area to flow area, vane orientation, and azimuthal extension of the vanes around the rod circumference. The relevant fluid parameters of the test data are mass velocity, quality, and pressure.

If the new geometry is not similar to or bracketed by the geometry and fluid parameters of the test data, W will evaluate the geometry relative to the existing test data by either using additional test data (either using existing DNB data or performing a bundle test to bracket the new geometry) or by comparing it against other NRC approved correlation(s) which are valid over the geometric or fluid parameter range of interest for the new design.

If additional test data are used, statistical tests will be performed to show that the new data are from the same population as the existing data base. The new data will then be pooled with the existing data base and a new limit DNB ratio will be recalculated based on the statistics of the pooled data base.

If the new data cannot be shown to be from the same population, they may be treated explicitly as a separate population using the existing DNB correlation; or a modified or new DNB correlation may be developed. If this step is necessary, this would constitute a submittal for NRC review and approval. W was questioned that if old test data are used or if new thermal hydraulic tests are performed, what specific statistical tests will be used to determine if the CHF test data for the new geometric design are from the same population as the data base from an existing DNB correlation? W's response and definition of statistical tests are detailed in References 3 and 5. These statistical tests follow those done for comparison of data to previously approved DNB correlations as done in Reference 26. PNL concludes that these procedures and statistical tests for determining if a NRC review is required are acceptable.

4.4 OVERHEATING OF FUEL PELLETS

Bases/Criteria - As a second method of avoiding cladding failure due to overheating, W precludes centerline pellet melting during normal operation and AOOs (References 10, 13, 15, 16, 17, and 18). This design limit is the same as given in the SRP and has been approved for application to current W fuel designs up to a rod-average burnup level of 60 GWd/MTM. In order to ensure that this basis is met, W imposes a design limit on fuel temperatures such that there is a 95% probability that the peak linear heat generation rate rod will not exceed the fuel melting temperature. W has placed a temperature limit on fuel melting at extended fuel burnup levels that is considered to be conservative. Therefore, PNL concludes that W's design limit for fuel melting is acceptable for application to W fuel design changes up to the approved rod-average burnup limit of 60 GWd/MTM.

Evaluation - W uses either the PAD 3.3 or 3.4 codes to evaluate fuel melting (References 13 and 27). Both of these codes have been approved for evaluation of W designs, however, PNL recommends that the PAD 3.4 code be used because it is the newer and more thoroughly verified of the two codes. Therefore, PNL concludes that W's analysis methods for fuel melting are acceptable for application to W fuel design changes up to the approved rod-average burnup limit of 60 GWd/MTM.

4.5 PELLET/CLADDING INTERACTION

Bases/Criteria - As indicated in Section 4.2 of the SRP, there are no generally applicable criteria for pellet/cladding interaction (PCI) failure. However, two acceptance criteria of limited application are presented in the SRP for PCI: 1) less than 1% transient induced cladding strain; and 2) no centerline fuel melting. Both of these limits have been adopted by W for use in evaluating their fuel designs (References 10, 13, 15, 16, 17, and 18) and have been approved by the NRC for application to a rod-average burnup level of 60 GWd/MTM. PNL concludes that these limits are also acceptable for application to W fuel design changes up to a rod-average burnup level of 60 GWd/MTM.

Evaluation - W uses the PAD 3.4 code to show that their fuel meets both the cladding strain and fuel melt criteria as discussed in Sections 3.2 and 4.4, respectively. As noted earlier, PNL concludes that this code is acceptable for application to current W fuel design changes up to a rod-average burnup limit of 60 GWd/MTM.

From the above, PNL concludes that W analysis methods adequately address the effects of PCI for future fuel design changes.

4.6 CLADDING RUPTURE

Bases/Criteria - There are no specific design limits associated with cladding rupture other than the 10 CFR 50, Appendix K (Reference 28) requirement that the degree of swelling not be underestimated. The W rupture model is an integral portion of the W emergency core cooling system (ECCS) evaluation model for determining the peak cladding temperature (PCT). The W design basis also states that the degree of cladding swelling or ballooning not be underestimated. Therefore, PNL concludes that the W design basis is acceptable.

Evaluation - The cladding deformation and rupture models used by W in their LOCA-ECCS analysis are directly coupled to their models for cladding ballooning and flow blockage. A more detailed discussion of these models is provided in the section that addresses cladding ballooning and flow blockage (see Section 5.3).

4.7 FUEL ROD MECHANICAL FRACTURING

Bases/Criteria - The term "mechanical fracture" refers to a cladding defect that is caused by an externally applied force such as a load derived from core-plate motion or a hydraulic load. These loads are bounded by the loads of a safe-shutdown earthquake (SSE) and LOCA, and the mechanical fracturing analysis is usually done as a part of the SSE-LOCA loads analysis, see Section 5.4 of this TER.

Evaluation - The discussion of the SSE-LOCA loading analysis is given in Section 5.4 of this TER.

4.8 THERMOHYDRAULIC STABILITY

Bases/Criteria - The W design basis is to ensure that normal operation or AOOs will not lead to thermohydraulic instability in the reactor core. The design limits are that Ledinegg instability will not occur and that a large margin will exist to density wave dynamic instability. These design limits have been approved by the NRC for previous W fuel designs (References 15 and 16) and, therefore, PNL concludes that they are acceptable for application to W fuel design changes.

Evaluation - In order to prevent a Ledinegg instability, the slope of the reactor coolant system pressure drop, flow rate curve must be shown to be algebraically larger than the loop supply (pump head) flow rate curve. This is generally the case for a PWR and, therefore, this type of instability generally does not exist in a PWR. The margin to density wave stability is calculated based on flow stability experiments, the inherent thermal hydraulic characteristics of W designs and the analysis methods in Reference 28. These methods have been applied to previous W fuel designs (References 15 and 16) and, therefore, PNL concludes that they are acceptable for application to W fuel design changes.

5.0 FUEL COOLABILITY

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

5.1 FRAGMENTATION OF EMBRITTLED CLADDING

Bases/Criteria - The LOCA is the design basis event resulting in 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 limit the effects of cladding oxidation for a LOCA, W uses an acceptance criteria of 2200°F (1200°C) on peak cladding temperature and 17% on maximum cladding oxidation as prescribed by 10 CFR 50.46.

For events other than the LOCA, there are no separately established temperature or oxidation criteria. However, it is clear that for short-term events such as a locked rotor/shaft break accident, the 2200°F (1200°C) peak cladding temperature and 17 percent oxidation LOCA criteria are not really meaningful, because the temperature history for such an event is much shorter than that of a LOCA. For events such as a locked rotor accident, W uses a unique PCT criterion of 2700°F (1480°C).

The W 2700°F (1480°C) PCT limit was selected taking into consideration the short time (a few seconds) that the fuel is calculated to be in DNB for a locked rotor type event and the fact that the PCT and total metal-water reaction at the fuel hot spot is not expected to impact fuel coolable geometry. The NRC has approved (Reference 15) the 2700°F (1480°C) PCT limit for short-term undercooling events such as locked rotor as an acceptable coolability limit for W fuel designs up to current burnup levels. PNL concludes that this coolability limit is also acceptable for application to W fuel design changes.

Evaluation - The Baker-Just equation for the Zircaloy-4 water reaction rate is used by W to determine the amount of cladding oxidation for Zircaloy-4 and ZIRLO during a LOCA. The Baker-Just equation is prescribed in 10 CFR 50, Appendix K (Reference 29). Therefore, PNL concludes that the use of the Baker-Just equation for ZIRLO oxidation during a LOCA for W fuel design changes is acceptable.

W utilizes the PAD 3.4 code to provide input of initial stored energy to the LOCA analysis and PNL concludes that this code is also acceptable for application to fuel design changes.

System transients and fuel rod thermal transients for the locked rotor/shaft break accident are analyzed using codes and methods which have either been approved by the NRC or accepted by the NRC in specific reviews (References 30, 31, and 32). PNL concludes that these codes and methods are also acceptable for application to W fuel design changes.

5.2 VIOLENT EXPULSION OF FUEL

Bases/Criteria - In a severe reactivity initiated accident (RIA) such as a control rod 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 an RIA event, Regulatory Guide 1.77 recommends that the radially averaged energy deposition at the hottest axial location be restricted to less than 280 cal/g.

There is evidence that the radial average enthalpy limit of 280 cal/g UO_2 may not maintain core coolability, particularly for irradiated fuel rods (Reference 33). These investigators have further indicated that "neither severe fuel rod damage not loss of normal geometry is expected at radial average peak fuel enthalpies below about 240 cal/g UO_2." These investigators also indicated that while irradiated fuel rods failed at significantly lower enthalpies than unirradiated rods, there was little sensitivity to the degree of burnup above approximately 33 GWd/MTM. However, it was noted that the number of previously irradiated rods, i.e., rods with burnup levels greater than 4 GWd/MTM, was very small having only four rods with burnup levels greater than 10 GWd/MTM and two above 30 GWd/MTM. The two rods with burnup levels above 30 GWd/MTM had relatively benign failures with radial average fuel enthalpies of approximately 150 cal/g UO_2.

The W design limit for the average fuel pellet enthalpy is 200 cal/g (360 Btu/lbm) for irradiated and unirradiated fuel (Reference 1). W's design limit is more conservative than the 280 cal/g limit given in Regulatory Guide 1.77 and has been previously approved in the review of WCAP-7588 (Reference 34) and in W's submittal to extend their rod-average burnup level to 60 GWd/MTM (Reference 13). W fuel design changes are not expected to impact the fuel enthalpy limit. Therefore, PNL concludes that W design limits for fuel dispersal are acceptable for application to W fuel design changes.

Evaluation - The W analysis methods for RIA events are not impacted by W fuel design changes and, therefore, remain acceptable for application to fuel design changes.

5.3 CLADDING BALLOONING AND FLOW BLOCKAGE

Bases/Criteria - Zircaloy cladding will balloon (swell) under certain combinations of temperature, heating rate, and stress during a LOCA. There are no specific design limits associated with cladding ballooning other than the 10 CFR 50, Appendix K (Reference 29) requirement that the degree of swelling not be underestimated. The W design limits state that the models utilize applicable test data in such a way as to properly estimate the prerupture clad strain, the rupture (burst) strain at the location of clad rupture and not underestimate the assembly flow blockage. These design limits consist of the Appendix K requirement and, therefore, PNL concludes they are acceptable for application to W fuel design changes.

Evaluation - The W cladding ballooning and flow blockage model is directly coupled to the cladding rupture temperature model for the LOCA-ECCS analysis and these are addressed in approved References 35, 36, 37, and 38. PNL concludes that these models are consistent with those in NUREG-0630 (Reference 39) and, therefore, remain acceptable for application to W fuel design changes.

5.4 FUEL ASSEMBLY STRUCTURAL DAMAGE FROM EXTERNAL FORCES

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

The W design basis is that the fuel assembly will maintain a geometry that is capable of being cooled under the worst case design basis accident and that no interference between control rods and thimble tubes will occur during a SSE. This is nearly identical to the design basis presented in the SRP and, therefore, PNL concludes that this basis is acceptable for application to W fuel design changes.

Evaluation - Generic analysis methods for performing combined LOCA-seismic loading analyses have been described by W in WCAP-9401-P-A (and WCAP-9402-A) (Reference 26). These analysis methods not only include the fuel assembly structural response, but also fuel rod cladding loads. These methods have been approved by the NRC and, therefore, PNL concludes they remain acceptable for application to W fuel design changes.

It should be noted that in WCAP-12488, it is stated that "best estimate" models may be used to determine spacer grid loads. W was questioned if these new "best estimate" models have been approved (Reference 2). W responded in Reference 3 that they have not been approved because they are still under NRC review. W has further stated that if they apply "best estimate" technology to grid deformation/core coolability analyses, these will be submitted to NRC as part of a plant's licensing application.

6.0 LEAD TEST ASSEMBLIES

Lead test assemblies (LTAs) are inserted in PWR cores to obtain early irradiation experience on new product features subjected to normal operating conditions. Several guidelines associated with the LTA demonstration programs have been accepted by the NRC in the past and include: 1) the design of the LTAs are mechanically and hydraulically compatible with existing fuel assemblies, 2) the peaking factors meet the Technical Specification limits, 3) NRC approved/accepted safety/design evaluation methodology and codes described in Section 4.0 of Reference 1 are used, 4) no SAFDLs described in Section 4.0 of Reference 1 are exceeded, 5) not more than eight LTAs per core are normally inserted, and 6) the LTAs are not to be the limiting assembly in the core.

PNL concludes that LTAs inserted in a reactor core do not need to obtain prior NRC approval if the review of the reload core results, including LTAs, meet the guidelines stated above.

7.0 CONCLUSIONS

PNL has reviewed the design criteria and evaluation methods in WCAP-12488 with respect to making fuel design changes and changes to W's analytical models without NRC review and approval. The design criteria and evaluation methods as defined in WCAP-12488 and References 3 and 5 are found to be acceptable up to current rod-average burnup limits of 60 GWd/MTM. However, W has agreed to submit fuel design and analytic model changes that are instituted under WCAP-12488 to NRC for information purposes. It is up to W to determine that the design criteria and analytical methods in WCAP-12488 are applicable to specific design changes. For each application of the "Fuel Criteria Evaluation Process," W must demonstrate compliance to this process and should provide this information to the NRC staff. W should also understand that future audits are possible based on this information.

It is also noted that the extended burnup section of WCAP-12488 was withdrawn by W and, therefore, was not reviewed nor approved as a part of WCAP-12488.

8.0 REFERENCES

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4. Letter, from N. J. Liparulo (W) to R. C. Jones (NRC), dated July 17, 1992 (ET-NRC-92-3723), subject "Supplement to Additional Information on WCAP-12488, 'Westinghouse Fuel Criteria Evaluation Process'," Proprietary.

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SECTION C

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

April 2, 1990
NS-NRC-90-3482

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Attention: V. H. Wilson, Planning, Program and Management Support Branch

Subject: WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process" (Proprietary)

Dear Ms. Wilson:

Enclosed are twenty-five (25) copies of the Westinghouse proprietary topical report, WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process."

The enclosed report documents a process requested by your staff to develop criteria which would allow Westinghouse to modify their fuel designs without requiring NRC review. The concept is that once the criteria for the modifications are developed and approved by the NRC, subsequent fuel changes would not require NRC approval provided the criteria are met. This process of identifying and using criteria to avoid specific NRC review has been entitled the, "Westinghouse Fuel Criteria Evaluation Process" (WFCEP).

In the past, Westinghouse has continually modified their fuel designs due to fuel upgrades, design fixes, manufacturing enhancements, etc., using approved NRC methods. Once the enclosed criteria are approved, new Westinghouse fuel design changes satisfying the criteria would not require extensive staff review; therefore, a timely review and approval of this topical is requested.

This submittal contains Westinghouse proprietary information of trade secrets, commercial or financial information which we consider privileged or confidential pursuant to 10CFR9.5(4). Therefore, it is requested that the Westinghouse proprietary information attached hereto be handled on a confidential basis and be withheld from public disclosure.

This material is for your internal use only and may be used solely for the purpose for which it is submitted. It should not be otherwise used, dislcosed, duplicated, or disseminated, in whole or in part, to any other person or organization outside the Office of Nuclear Reactor Regulation without the express written approval of Westinghouse.

Correspondence with respect to any Application for Withholding should reference AW-90-002 and should be addressed to R. A. Wiesemann, Manager of Regulatory and Legislative Affairs, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Pursuant to the requirements of 10CFR170.12(c) enclosed is an application fee for $150.00.

Very truly yours,

W. J. Johnson, Manager
Nuclear Safety Department

cc: R. C. Jones, RSB

Westinghouse Electric Corporation
Energy Systems
Box 355
Pittsburgh Pennsylvania 15230-0355

April 2, 1990
AW-90-002

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Attention: V. H. Wilson, Planning, Program and Management Support Branch

Reference: Letter from W. J. Johnson to V. H. Wilson, NS-NRC-90-3482, dated April 2, 1990

Subject: WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process" (Proprietary)

Dear Ms. Wilson:

The above referenced letter contains information proprietary to the Westinghouse Electric Corporation.

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

Accordingly, we request that the material be treated as proprietary information within the provisions of 10CFR9.5(4), "Freedom of Information Act Regulations." If there is a need to make public disclosure of the material prior to a separate Westinghouse submittal for docket in accordance with the provisions of 10CFR2.790(a), please notify Westinghouse prior to making a disclosure determination. The proprietary material for which withholding is being required is of the same technical type as that proprietary material previously submitted as Affidavit AW-89-088.

Correspondence with respect to the proprietary aspects of this submittal should reference AW-90-002 and should be addressed to the undersigned.

Very truly yours,

Robert A. Wiesemann

Robert A. Wiesemann, Manager
Regulatory & Legislative Affairs

cc: K. Holzle, Esq.
Office of the General Counsel, NRC

SECTION D

WCAP - 12488

WESTINGHOUSE FUEL CRITERIA EVALUATION PROCESS

April 1990

S. L. Davidson, Editor

Approved: E H Novendstern
E. H. Novendstern, Manager
T&H Design and Fuel Licensing

Approved: M P Osborne
M. P. Osborne, Manager
Transient Analysis

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P. O. Box 3912
Pittsburgh, Pennsylvania 15230

ACKNOWLEDGEMENTS

The Editor would like to acknowledge the efforts of the many contributors from both the Westinghouse Commercial Nuclear Fuel Division and the Westinghouse Nuclear and Advanced Technology Division who helped develop the functional requirements and the design criteria for the process documented in this report.

ABSTRACT

In 1987 the NRC proposed that a process for evaluating new or modified fuel designs be developed by each fuel supplier. Once the process is approved, mechanical changes to the fuel assembly, which satisfy the criteria evaluated by the process, will not need NRC staff review prior to a PWR plant application.

This document outlines a process and the criteria by which new or modified fuel designs will be evaluated. Providing that no changes to the Technical Specifications will be required because of the fuel design change, the change may then be implemented without prior NRC review and approval, if it meets the criteria specified within this document. Further, for instances that require Technical Specification changes, only the Technical Specifications will need to be approved by the NRC; NRC review of the new design will not be required. This document complements the NRC approved Reload Safety Evaluation Methodology topical report, WCAP-9272-P-A, that focuses on the analytical and safety analysis aspects of the reload design activity.

The subsequent sections in this document describe the process and list the criteria for the Westinghouse Fuel Criteria Evaluation Process (FCEP).

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FUEL CRITERIA EVALUATION PROCESS

1.0 INTRODUCTION

Consistent with the growth in technology, Westinghouse frequently modifies fuel designs. Generally, the changes in design will utilize approved methods and thus the qualification and approval of the new designs or mechanical design change(s) follow a straightforward process. Rarely does the NRC review result in a significant change to the design or related issues. Because of this, the NRC in a November 3, 1987 NRC memo from A.C. Thadani to R. W. Starostecki. requested vendors to submit a set of criteria, to be satisfied by new fuel designs, Once the criteria are approved, new fuel designs or changes satisfying the criteria will not need explicit NRC staff review and reference can be made to this document.

This document describes the process, called by Westinghouse the Fuel Criteria Evaluation Process (FCEP), which results in a thorough evaluation of the appropriate criteria. The process assesses the safety significance and assures the licensability of PWR fuel system mechanical design change(s) by confirming that appropriate criteria related to the fuel system are satisfied. These criteria, which have been accepted by the NRC, are evaluated using NRC accepted codes and methods. This document applies to fuel manufactured by Westinghouse.

Implementation of this process will allow nuclear power plant licensees to make mechanical design changes without prior NRC approval, consistent with the provisions of 10CFR50.59, if these changes satisfy the design bases and limits described in this report. For instances where changes to the plant's Technical Specification(s) will be required, only the Technical Specification changes will need to be approved by the NRC; NRC review and approval of the design changes will not be required.

For the purposes of this document, the "Fuel System Design" is consistent with Section 4.2, "Fuel System Design " (Revision 3*) of the NRC Standard Review Plan (SRP NUREG-0800), i.e.,

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* For non-SRP licensed plants and for plants whose license is based on older versions of the SRP, this process is also applicable and will be applied consistent with the plant's specific licensing basis.

"The fuel system consists of arrays (assemblies or bundles) of fuel rods including fuel pellets, insulator pellets, springs, tubular cladding, end closures, hydrogen getters, and fill gas, of spacer grids and springs, of end plates; channel boxes; and reactivity control rods. In the case of the control rods, this section covers the reactivity control elements that extend from the coupling interface of the control rod drive mechanism into the core."

Westinghouse includes, but is not limited to, discrete and integral fuel burnable absorbers, water displacer rods, stainless steel or zircaloy replacement rods inserted in reconstituted fuel assemblies and hafnium power suppression rods, in the definition of the fuel system.

2.0 BACKGROUND

The fuel system design is based upon 10CFR50, Appendix A, "General Design Criteria;" specifically, by Criterion 10, "Reactor Design" which states:

> "The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specific acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences."

Section 4.2 of the Standard Review Plan (SRP) indicates that the NRC reviews the vendor-established specific acceptable fuel design limits to "... provide assurance that:

(a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences,

(b) fuel system damage is never so severe as to prevent control rod insertion when it is required,

(c) the number of fuel rod failures is not underestimated for postulated accidents, and

(d) coolability is always maintained."

Westinghouse has established and obtained NRC approval for the design bases for several PWR fuel system designs (e.g., WCAP-9500-A, "Reference Core Report - 17 x 17 Optimized Fuel Assembly," May 1982, WCAP 10444-P-A, "Westinghouse Reference Core Report - VANTAGE 5 Fuel Assembly," September 1985, and WCAP-10125-P-A, "Extended Burnup Evaluation of Westinghouse Fuel," December 1985). These design bases, or Specified Acceptable Fuel Design Limits (SAFDLs), and their respective evaluations comply with the "Acceptance Criteria" of SRP 4.2, "Fuel System Design," and as such provide assurance that the fuel system is mechanically designed to perform safely, consistent with General Design Criterion 10. The plant specific requirements outlined in the NRC's SERs for WCAP-9500-A and WCAP-10444-P-A will be adhered to in applying the FCEP to plant specific applications.

3.0 FUEL CRITERIA EVALUATION PROCESS DESCRIPTION/APPLICATION

3.1 Description

The FCEP allows for 10CFR50.59 conclusions to be reached by demonstrating that the criteria defined in this document are used for the evaluation of the fuel mechanical change(s) and are met. Where changes to plant Technical Specifications are required, an NRC review of the fuel mechanical change(s) will not be required, if the criteria are satisfied. Only the associated changes to the Technical Specifications will need to be reviewed by the NRC on a plant specific application.

The FCEP is consistent with and supplements the reload design and licensing approach described in the approved topical WCAP-9272-P-A. The bounding analysis concept for the safety analyses described in WCAP-9272-P-A remains unaffected. Also, when applied to specific design changes, the FCEP will show compliance with the SAR acceptance limits. The FCEP is also applicable when Westinghouse acts for a utility who provides portions of the calculations using similar computer codes and methodologies.

3.2 Application

For each fuel system design change, Westinghouse will review the design bases/limits given in Section 4.0 (in conjunction with the licensing basis of the specific plant) and evaluate the effect of the change relative to these bases. Design bases for the PWR safety analysis address fuel system damage mechanisms and provide limiting values for important parameters such that the predicted effects will be confined to acceptable limits.

As stated in the Section 1 of this report, the fuel criteria are evaluated using NRC-approved codes and methods. These codes and methods are referenced throughout the report, and the evaluations that are described are usually done solely by Westinghouse. However, some licensees, who have signed agreements to use Westinghouse design methods/codes, desire to do portions of the evaluation themselves. Upon the licensee obtaining the necessary NRC approval, they can utilize the methods/codes for the areas in which they have received NRC approval. Additionally, when the licensee is not using Westinghouse design codes/methods, the licensing applicability of codes and methodology used to model the fuel change is the responsibility of the licensee. Upon NRC approval of the licensee's non-Westinghouse neutronic methods/codes, results can be used in conjunction with the FCEP process to assure all criteria are satisfied.

The design failure point or system limitation at which failure would occur and beyond which operation would be unacceptable is often not known with complete certainty. The NRC defines acceptance limits that are more restrictive than the design failure points or system limitations to assure that these conditions are not reached. In this document, the acceptance limit, i.e., the value prescribed/approved/accepted by the NRC for Westinghouse application, is given for each design parameter. This limit, often called a design limit in NRC SERs, if exceeded for a proposed design change, might constitute grounds for the denial of a license or an amendment to the license. For certain evaluations, Westinghouse uses limits that are more restrictive than NRC approved acceptance limits; these are noted as appropriate in this document. For completeness this document addresses all applicable Standard Review Plan items.

For convenience, the NRC acceptance criteria for the design limits are grouped into three categories in the SRP. These categories are:

(a) fuel system damage criteria which are most applicable to Condition I (Normal Operation)

(b) fuel rod failure criteria, which apply to Conditions II, III and IV (Incidents of Moderate frequency, Infrequent Incidents, and Limiting Faults)

(c) fuel coolability criteria, which apply to Condition IV (Limiting Faults)

Section 4.0 describes the design bases and limits associated with the above categories of acceptance criteria and also the bases and limits for the nuclear design. Section 5.0 is an application of the Section 4.0 criteria to Westinghouse Lead Test Assemblies (LTAs). The FCEP is flexible to incorporate new or improved methods or codes once approved by the NRC; the method describing this is found in Section 6.0. Section 7.0 is a new Section and provides supplemental information for Section 6.0.

SECTION 4.0 FUEL DESIGN BASES AND LIMITS

This section of the report describes each of the criteria that will be evaluated when mechanical design changes to the fuel system are made. The associated design parameters, design bases, and design or acceptance limits, along with the method and associated references that Westinghouse uses for evaluation are summarized. The definition of these items are:

Design Category - The NRC groups acceptance criteria for design limits into three categories in the Standard Review Plan. (See Section 3.2)

Design Parameter - The specific parameter that will be evaluated.

Design Basis - The reason that the design parameter needs to be considered in the safety evaluations.

Acceptance Limit - The acceptance limit is the value prescribed/approved/accepted by the NRC. For certain evaluations, Westinghouse uses limits that are more restrictive than NRC approved acceptance limits; these are described as appropriate.

Design Evaluation - This is a brief description of the methods used to evaluate the design or acceptance limits. Often, the evaluation methods are described more fully in Westinghouse topical reports that have been approved by the NRC. These methods are referenced in this section.

References(s) - A list of topical reports that are applicable to the design evaluation.

Plant Conditions - ANSI Committee Project ANS N18.2 defines the following categories for plant conditions:

Condition I	Normal Operation
Condition II	Incidents of Moderate Frequency
Condition III	Infrequent Incidents
Condition IV	Limiting Faults

Design Category - Fuel System Damage and Fuel Rod Failure Criteria

Design Parameter - Clad Stress

Design Basis - The fuel system will not be damaged due to excessive fuel clad stress.

Acceptance Limit - The volume average effective stress calculated with the Von Mises equation considering interference due to uniform cylindrical pellet-clad contact, caused by pellet thermal expansion, pellet swelling and uniform clad creep, and pressure differences, is less than the 0.2 percent offset yield stress with due consideration to temperature and irradiation effects under Condition I and II events, Reference 1. While the clad has some capability for accommodating plastic strain, the yield stress has been accepted as a conservative design limit.

Design Evaluation - The Westinghouse fuel performance code described in Reference 2 is used for evaluating clad stress limits.

References(s)

1. Davidson, S. L. (ed.) et al., "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125-P-A, December 1985.

2. Weiner, R. A., et al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851-P-A, August 1988.

Design Category - Fuel System Damage and Fuel Rod Failure Criteria

Design Parameter - Clad Strain

Design Basis - The fuel system will not be damaged due to excessive fuel clad strain.

Acceptance Limit - The total plastic tensile creep strain due to uniform clad creep and uniform cylindrical fuel pellet expansion associated with fuel swelling and thermal expansion is less than 1% from the unirradiated condition, Reference 1. The acceptance limit for fuel rod clad strain during Condition II events is that the total tensile strain due to uniform cylindrical pellet thermal expansion is less than 1% from the pre-transient value. These limits are consistent with proven practice.

Design Evaluation - The Westinghouse fuel performance code described in Reference 2 is used to evaluate margin to clad strain limits.

Reference(s)

1. Davidson, S. L. (ed.) et al., "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125-P-A, December 1985.

2. Weiner, R. A., et al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851-P-A, August 1988.

Design Category - Fuel System Damage and Fuel Rod Failure Criteria

Design Parameter - Clad Fatigue

Design Basis - The fuel system will not be damaged due to fatigue.

Acceptcnce Limit - The fatigue life usage factor is less than 1.0, Reference 1. That is, for a given strain range, the number of strain fatigue cycles are less than those required for failure, considering a minimum safety factor of 2 on the stress amplitude or a minimum safety factor of 20 on the number of cycles, whichever is more conservative.

Design Evaluation - The evaluation of the fatigue limit assumes conservative load follow scenarios over the life of the fuel rod. The Westinghouse fuel performance code, Reference 2, is used to determine the strain range for the fatigue life usage analysis. The Langer - O'Donnell fatigue model, Reference 3, constitutes the basic approach taken in the fatigue analysis, with the empirical factors of their correlation modified in order to conservatively bound the results of Westinghouse testing programs. Westinghouse fatigue test programs and the resulting data on which the design fatigue life is based are presented in Reference 4. The design equations follow the concept for the fatigue design criterion described in the ASME Boiler and Pressure Vessel Code, Section III.

Reference(s)

1. Davidson, S. L. (ed.) et al., "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125-P-A, December 1985.

2. Weiner, R. A., et al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851-P-A, August 1988.

3. O'Donnell, W. J. and Langer, B. F., "Fatigue Design Basis for Zircaloy Components," Nuclear Science and Engineering, 20, 1-12, 1964.

4. Davidson, S. L. and Iorii, J. A., "Reference Core Report 17 x 17 Optimized Fuel Assembly," WCAP-9500-A, May 1982.

Design Category - Fuel System Damage and Fuel Rod Failure Criteria

Design Parameter - Clad Oxidation

Design Basis - The fuel system will be not be damaged due to excessive fuel clad oxidation.

Acceptance Limit - The calculated clad temperature (metal-oxide interface temperature) will be less than [ ]$^{+}$ during steady state operation, Reference 1. For Condition II events, the calculated clad temperature will not exceed [ ]$^{+}$. $^{+}$(a,c) $^{+}$(a,c)

Design Evaluation - The Westinghouse fuel performance code described in Reference 2 is used to evaluate clad surface temperature limits.

Reference(s)

1. Davidson, S. L. (ed.), et al., "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125-P-A, December 1985.

2. Weiner, R. A., et al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851-P-A, August 1988.

Design Category - Fuel System Damage and Fuel Rod Failure Criteria

Design Parameter - Zircaloy Clad Hydrogen Pickup

Design Basis - The fuel system will be operated to prevent significant degradation of mechanical properties of the clad at low temperatures, as a result of hydrogen embrittlement caused by the formation of zirconium hydride platelets.

Acceptance Limit - The hydrogen pickup level in the zircaloy clad will be less than or equal to [ ]$^{+}$ ppm at the end of fuel operation, References 1 and 2. +(a,c)

Design Evaluation - As described in References 1 and 2, compliance with the hydrogen pickup limits has been demonstrated based on measured hydrogen pickup values obtained on fuel rods operated to burnups exceeding the current rod average burnup limit.

Reference(s)

1. Davidson, S. L. (ed.), et al., "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125-P-A, December 1985.

2. Weiner, R. A., et al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851-P-A, August 1988.

Design Category - Fuel System Damage and Fuel Rod Failure Criteria

Design Parameter - Fuel Rod Axial Growth

Design Basis - The fuel rods will be designed with adequate clearance between the fuel rod ends and the top and bottom nozzles to accommodate the differences in the growth of the fuel rods and the growth of the fuel assembly.

Acceptance Limit - The Westinghouse design limit for fuel rod growth, References 1 and 2, is that no interference between the fuel rods and the fuel assembly top and bottom nozzles will occur.

Design Evaluation - The fuel rod growth model described in Reference 3 is used to evaluate fuel rod growth limits. The growth of the annealed zircaloy thimble tubes is small relative to the growth of the cold-worked clad resulting in a net reduction of the rod-to-nozzle growth gap as a function of increased exposure.

Reference(s)

1. Davidson, S. L. and Iorii, J. A., "Reference Core Report 17x17 Optimized Fuel Assembly," WCAP-9500-A, May 1982.

2. Davidson, S. L. (ed.), et al., "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125-P-A, December 1985.

3. Weiner, R. A., et al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851-P-A, August 1988.

Design Category - Fuel System Damage and Fuel Rod Failure Criteria

Design Parameter - Clad Flattening

Design Basis - Fuel rod failures will not occur due to clad flattening.

Acceptance Limit - The core residence time will not exceed the calculated time which corresponds to a flattened rod frequency of 1.0, References 1 and 2.

Design Evaluation - Evaluation of fuel rod clad flattening is performed using the clad flattening model in Reference 1. Calculations performed for current generation stable fuel in pressurized rods show that the predicted clad flattening time exceeds the expected residence time.

Reference(s)

1. George, R. A., et al., "Revised Clad Flattening Model," WCAP-8381, July 1974.

2. Davidson, S. L. (ed.), et al., "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125-P-A, December 1985.

Design Category - Fuel System Damage and Fuel Rod Failure Criteria

Design Parameter - Rod Internal Pressure

Design Basis - The fuel system will not be damaged due to excessive fuel rod internal pressure.

Acceptance Limit - The internal pressure of the lead rod in the reactor will be limited to a value below that which could cause the diametral gap to increase due to outward clad creep during steady state operation or for extensive DNB propagation to occur, Reference 1.

Design Evaluation - The fuel performance code, Reference 2, is used to evaluate rod internal pressure as a function of irradiation time and fuel duty. Rod internal pressure at steady state conditions is used in the LOCA analyses to show compliance to 10CFR50.46.

Reference(s)

1. Risher, D. H. (Editor), "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis," WCAP-8963-P-A, August 1978.

2. Weiner, R. A., et. al, "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851-P-A, August 1988.

Design Category - Fuel System Damage and Fuel Rod Failure Criteria

Design Parameter - Fuel Clad Fretting Wear

Design Basis - The fuel system will not be damaged due to fuel clad fretting.

Acceptance Limit - Westinghouse uses a wall thickness reduction of [ ]$^+$ percent, Reference 1, as a general guide in evaluating clad imperfections, including fretting wear marks. Furthermore, both clad stress and fatigue acceptance limits must be met. $^+$(a,c)

Design Evaluation - The fretting wear evaluation is performed: (1) on the basis of experimental data [ ]$^+$ or (2) using an [$^+$(a,b,c)

]$^+$ flow tests, References $^+$(a,b,c)

2 and 3, or (3) vibration [

]$^+$ and/or (4) using the calculated [$^+$(a,c)

]$^+$ $^+$(a,c)

Reference(s)

1. Davidson, S. L. (ed.) et al., "Extended Burnup Evaluation of Westinghouse Fuel, "WCAP-10125-P-A, December 1985.

2. Davidson, S. L., and Iorii, J. A., "Reference Core Report 17x17 Optimized Fuel Assembly," WCAP-9500-A, May 1982.

3. Davidson, S. L., Kramer, W. R. (Eds.) "Reference Core Report VANTAGE 5 Fuel Assembly," WCAP-10444-P-A, September 1985.

Design Category - Fuel System Damage and Fuel Rod Failure Criteria

Design Parameter - Fuel Rod Clad Rupture (Burst)

Design Basis - Appendix K to 10CFR50 requires that an accurate model be used in LOCA analyses to predict rupture (burst) of the fuel rod clad.

Acceptance Limit - The model uses appropriate test data. It includes the effects of differential pressure, local temperature, and heat up rate. It is used in the Appendix K to 10CFR50 evaluation of PCT.

Design Evaluation - The large break LOCA evaluation, References 1, 2, 3 and 4, uses the model described in Reference 5. The small break LOCA evaluation , References 6, and 7, uses a slightly different model which is documented in Reference 8.

References(s)

1. "Westinghouse ECCS Evaluation Model 1981 Version," WCAP-9220-P-A Revision 1, February 1982.

2. Young, M. Y., et al., "BART-A: A Computer Code for the Best Estimate Analysis of Reflood Transients," WCAP-9561-P-A, March 1984.

3. Besspiata, J. J., et al., "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," WCAP-10266-P-A, Revision 2, with Addenda, March 1987.

4. Hochreiter, L. E., et al., "Westinghouse Large Break LOCA Best Estimate Methodology," WCAP-10924-P-A Rev. 1, December 1988.

5. Powers, D. A. and Meyer, R. O., "Cladding Swelling and Rupture Models for LOCA Analysis," USNRC Report NUREG-0630, April 1980.

6 Lee, N., et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A, August 1985.

7. Meyer, P. E., "NOTRUMP, A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A, August 1985.

8. Skwarek, R., et al., "Westinghouse Emergency Core Cooling System Small Break, October 1975 Model," WCAP-8970-P-A, January 1979.

Design Category - Fuel System Damage and Fuel Rod Failure Criteria

Design Parameter - Fuel Pellet Overheating

Design Basis - The fuel rods will not fail due to fuel centerline melting for Condition I and Condition II events.

For the Condition IV Control Rod Ejection Accident, the fission product release from fuel rods experiencing centerline melting is accounted for in the radiological dose calculations.

Acceptance Limit - The Westinghouse design limit for fuel temperature analyses during Condition I and Condition II events is that there is at least a 95% probability that the peak kw/ft fuel rods will not exceed the UO_2 melting temperature, References 1 and 2. The melting temperature of unirradiated UO_2 is taken as 5080°F, decreasing by 58°F per 10,000 MWD/MTU exposure. A centerline fuel temperature limit of 4700°F has been selected by Westinghouse as the overpower limit used in design, References 3 and 4.

The Westinghouse design limit for the Condition IV, Control Rod Ejection Accident is that fuel pellet melting shall be limited to less than the innermost 10 percent of the pellet volume at the hot spot, References 5 and 6. This limit is consistent with assumptions used in the radiological dose calculations.

Design Evaluation - Fuel temperature evaluations for Condition I and Condition II events are performed using the fuel performance codes in References 7 and 8.

For the Condition IV Control Rod Ejection Accident, fuel melting is calculated using codes and methods which have either been approved by the NRC, References 5, 9 and 10, or accepted by the NRC in specific reviews.

Reference(s)

1. Davidson, S. L. and Iorii, J. A., "Reference Core Report 17x17 Optimized Fuel Assembly," WCAP-9500-A, May 1982.

2. Davidson, S. L., Kramer, W. R. (Eds.) "Reference Core Report VANTAGE 5 Fuel Assembly," WCAP-10444-P-A, September 1985.

3. Ellenberger, S. L., et al., "Design Bases for the Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Functions," WCAP-8745-P-A, September 1986.

4. Davidson, S. L. (ed.) et al., "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125-P-A, December 1985.

5. Risher, D. H., Jr., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," WCAP-7588, Revision 1-A, January 1975.

6. Letter from W. J. Johnson (W) to R. C. Jones (NRC), "Use of 2700°F Acceptance Limit in Non-LOCA Accidents," NS-NRC-89-3466, October 19, 1989.

7. Weiner, R. A., et al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851-P-A, August 1988.

8. Leech, W. J. et al., "Revised PAD Code Thermal Safety Model, WCAP-8720, Addenda 2, October 1982.

9. Risher, D. H., Jr. and R. F. Barry, "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979-P-A, January 1975.

10. Hargrove, H. G., "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

Design Category - Fuel System Damage and Fuel Rod Failure Criteria

Design Parameter - Non-LOCA Fuel Clad Temperature

Design Basis - The fuel rod damage due to overheating of the clad will be avoided for Condition I and Condition II events.

For postulated non-LOCA accidents (Condition III and Condition IV events) fuel rods assumed to fail are included in the radiological dose calculations.

Acceptance Limit - The DNB design criterion will be satisfied, thus avoiding fuel rod failure for: Condition I and II events, the Condition III complete Loss of Flow Event, and the Condition IV Main Steam Line Break Accident.

For Condition III and IV events in which limited DNB is predicted, the total number of fuel rods that exceed the criterion are assumed to fail for radiological dose calculation purposes, Reference 6.

Design Evaluation - System and core transients and fuel rod thermal transients for anticipated operational occurrences and postulated accidents are analyzed using codes and methods which have either been approved by the NRC or accepted by the NRC in specific reviews, References 1, 2, 3, 4 and 5.

Reference(s)

1. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A, April 1984.

2. Risher, D. H., Jr. and R. F. Barry, "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979-P-A, January 1975.

3. Hargrove, H. G., "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

4. Chelemer, H., et al., "THINC-IV - An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores," WCAP-8054-P-A, February 1989.

5. Risher, D. H., Jr., "An Evaluation of the Rod Ejection Accident In Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," WCAP-7588, Revision 1-A, January 1975.

6. Letter from W. J. Johnson (W) to R. C. Jones (NRC), "Use of 2700°F Acceptance Limit in Non-LOCA Accidents," NS-NRC-89-3466, dated October 19, 1989.

Design Category - Fuel System Damage and Fuel Rod Failure Criteria

Design Parameter - LOCA Fuel Clad Temperature

Design Basis - Adequate core cooling will be provided in all modes of operation and for Condition II, III and IV events to assure that the clad temperature, oxidation and embrittlement limits are not exceeded.

Acceptance Limit - Clad temperature is less than 2200°F for postulated LOCAs and a local clad oxidation limit is less than 17 percent of the total clad thickness prior to oxidation, References 1 thru 5. These limits also protect against embrittlement of the fuel clad.

Design Evaluation - LOCA evaluation models given in References 1 thru 6 are used to calculate the clad temperature transient for a hypothetical LOCA. The results of this analysis must show that the above peak clad temperature and local oxidation acceptance limits remain below that specified in 10CFR50.46. Any change to the calculated Peak Clad Temperature (PCT), as a result of a fuel design modification or upgrade, will be tracked and reported to the affected utilities for their use in meeting the reporting requirements of 10CFR50.46, as amended in 1988, and the reporting requirements of 10CFR50.59, and 10CFR50.71(e).

Reference(s)

1. "Westinghouse ECCS Evaluation Model 1981 Version," WCAP-9220-P-A Revision 1, February 1982.

2. Young, M. Y., et al., "BART-A: A Computer Code for the Best Estimate Analysis of Reflood Transients," WCAP-9561-P-A, March 1984.

3. Besspiata, J. J., et al., "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," WCAP-10266-P-A, Revision 2, with Addenda, March 1987.

4. Hochreiter, L. E., et al., "Westinghouse Large Break LOCA Best Estimate Methodology," WCAP-10294-P-A Rev. 1., December 1988.

5. Lee, N., et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081-A (Non-Proprietary), August 1985.

6. Meyer, P. E., "NOTRUMP, A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A (Proprietary) and WCAP-10080-P-A (Non-Proprietary), August 1985.

Design Category - Fuel System Damage and Fuel Rod Failure Criteria

Design Parameter - DNB

Design Basis - There will be at least a 95 percent probability at a 95 percent confidence level that DNB will not occur on the limiting fuel rods during Condition I and II events.

Acceptance Limit - The DNBR acceptance limit is 1.30 for the W-3 correlation above 1000 psia, References 1 to 4, 1.45 for the W-3 correlation in the range 500-1000 psia, Reference 5, 1.28 for the R-grid W-3 correlation, References 6, 7 and 8, 1.24 for the L-grid W-3 correlation, References 9 and 10, 1.17 for the WRB-1 correlation, Reference 11, and 1.17 for the WRB-2 correlation, Reference 12.

Design Evaluation - The THINC codes, References 13 to 16, are used to perform thermal-hydraulic analyses of the core and to calculate the minimum DNBR values in the hot channels. The WESTAR code, Reference 17, can also be used for these analyses. In the standard thermal design procedure, plant uncertainties are included in the thermal-hydraulic core analysis. In the Improved Thermal Design Procedure (ITDP, Reference 18), uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95 percent probability that the minimum DNBR will be greater than or equal to the correlation limit. In the Revised Thermal Design Procedure (RTDP, Reference 19), the plant and correlation uncertainties are combined statistically rather than directly as is done with ITDP. In the mini-RTDP, Reference 20, a subset of the RTDP design parameters in the determination of the DNBR uncertainty factor are used.

References

1. Tong, L. S., "DNB Prediction for an Axially Non-Uniform Heat Flux Distribution," WCAP-5584, Rev. 1, April 1966.

2. Tong, L. S., AEC Critical Review Series, "Boiling Crisis and Critical Heat Flux," TID-25887, August 1972.

3. Tong, L. S., et al., "Influence of Axially Non-uniform Heat Flux on DNB," Chemical Engineering Progress Symposium Series Vol. 62, No. 64, 1966.

4. Rosal, E. R., et al., "High Pressure Rod Bundle DNB Data with Axially Non-uniform Heat Flux," Nuclear Engineering and Design, Vol. 31, No. 1, pp 1-20, November 1974.

5. Letter from A. C. Thadani (NRC) to W. J. Johnson (Westinghouse), January 31, 1989, Subject: Acceptance for Referencing of Licensing Topical Report, WCAP-9226-P/ WCAP-9227-NP, "Reactor Core Response to Excessive Secondary Steam Releases."

6. Motley, F. E. and Cadek, F. F., "DNB Test Results for R Grid Thimble Cold Wall Cells," WCAP-7695-P-A Addendum 1, October 1972.

7. Hill, K. W., et al., "Effect of 17x17 Fuel Assembly Geometry on DNB," WCAP-8296-P-A, February 1975.

8. Motley, F. E., et al., "Critical Heat Flux Testing of 17x17 Fuel Assembly Geometry with 22-inch Grid Spacing," WCAP-8536, May 1975.

9. Motley, F. E. and Cadek, F. F., "Application of Modified Spacer Factor to L Grid Typical and Cold Wall Cell DNB," WCAP-7988-P-A, January 1975.

10. Motley, F. E. and Cadek, F. F., "DNB Test Results for New Mixing Vane Grids (R)," WCAP-7695-L, July 1972.

11. Motley, F. E., et al., "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," WCAP-8762-P-A, July 1984.

12. Davidson, S. L. and Kramer, W. R. (Ed.) "Reference Core Report - VANTAGE 5 Fuel Assembly," WCAP-10444-P-A, September 1985.

13. Chelemer, H., "THiNC-I, A Steady State Thermal-Hydraulic Interaction Code for Digital Computers," WCAP-2581, February 1964.

14. Shefcheck, J., "Application of the THINC Program to PWR Design," WCAP-7359-L, August 1969.

15. Chelemer, H., Chu, P. T., and Hochreiter, L. E., "THINC-IV - An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores," WCAP-8054-P-A, February 1989.

16. Hochreiter, L. E. and Chelemer, H., "Application of the THINC-IV Program to PWR Design," WCAP-8054-P-A, February 1989.

17. Ho, S. A., Olson, C. A. and Paik, I. K., "WESTAR: An Advanced Three-Dimensional Program for Thermal-Hydraulic Analysis of Light Water Reactor Cores," June 1988.

18. Chelemer, H., Boman, L. H., and Sharp, D. R., "Improved Thermal Design Procedure," WCAP-8567-P-A, February 1989.

19. Friedland, A. J. and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A, April 1989.

20. Ray, S., "MINI Revised Thermal Design Procedure (MINI RTDP)," WCAP-12178-P-A, October 1989.

Design Category - Fuel System Damage and Fuel Rod Failure Criteria

Design Parameter - Fuel Assembly Holddown Force

Design Basis - The fuel assembly will not be allowed to lift due to flow during normal operating conditions.

Acceptance Limit - The Westinghouse design limit is that the fuel assembly is designed to remain in contact with the lower core plate under all Condition I and II events with the exception of the turbine overspeed transient associated with a loss of external load, References 1 and 2.

Design Evaluation - The net upward force exerted on the fuel assembly is the result of the axial flow interacting with resistances along the flow path within a control volume. The pressure loss coefficients used to calculate the upward force are determined from hydraulic tests, References 1 and 2. Best estimate hydraulic force at normal operating conditions is calculated considering the best estimate flow and the best estimate core bypass flow with associated uncertainties.

Reference(s)

1. Davidson, S. L. and Iorii, J. A., "Reference Core Report - 17x17 Optimized Fuel Assembly," WCAP-9500-A, May 1982.

2. Davidson, S. L. and Kramer, W. R. (Eds.), "Reference Core Report VANTAGE 5 Fuel Assembly," WCAP-10444-P-A, September 1985.

Design Category - Fuel System Damage and Fuel Rod Failure Criteria

Design Parameter - Thermohydrodynamic Stability

Design Basis - Operation under Condition I and II events will not lead to thermohydrodynamic instability in the reactor core. The types of instability considered are Ledinegg or flow excursion static instability and density wave dynamic instability.

Acceptance Limit - The Westinghouse design limits for thermohydrodynamic stability are that Ledinegg instability will not occur and that a large margin will exist to density wave instability, References 1 and 2.

Design Evaluation - To prevent a Ledinegg instability, the slope of the reactor coolant system pressure drop - flow rate curve must be shown to be algebraically larger than the loop supply (pump head) pressure drop - flow rate curve. The margin to density wave instability is predicted by using the method of Ishii, Reference 3 . Typically increases on the order of 100% or greater of rated reactor power should be required for predicted inception of this type of instability.

Reference

1. Davidson, S. L. and Iorii, J. A., "Reference Core Report - 17x17 Optimized Fuel Assembly," WCAP-9500-A, May 1982.

2. Davidson, S. L. and Kramer, W. R. (Eds.), "Reference Core Report VANTAGE 5 Fuel Assembly," WCAP-10444-P-A, September 1985.

3. Saha, P., Ishii, M., and Zuber, N., "An Experimental Investigation of the Thermally Induced Flow Oscillations in Two-Phase Systems," V.01 Heat Transfer, November 1976, p. 616-622.

Design Cctegory - Fuel Coolability Criteria

Design Parameter - Clad Embrittlement During Locked Rotor/Shaft Break Accident

Design Basis - Clad embrittlement as a result of a postulated Locked Rotor/Shaft Break Accident will be avoided.

Acceptance Limit - The Westinghouse design limit for peak clad temperature following a Locked Rotor/Shaft Break Accident is 2700°F, References 1 and 2, and the limit for maximum clad oxidation is 17 percent.

Design Evaluation - System transients and fuel rod thermal transients for the Locked Rotor/Shaft Break Accident are analyzed using codes and methods which have either been approved by the NRC or accepted by the NRC in specific reviews, References 3, 4 and 5. This evaluation ensures that clad embrittlement is avoided and fuel coolability is maintained.

References

1. NRC SER, "Safety Evaluation of the Westinghouse Electric Corporation Topical Report WCAP-9500, Reference Core Report 17x17 Optimized Fuel Assembly," letter from Mr. R. L. Tedesco, NRC Division of Licensing, to Mr. T. M. Anderson, Westinghouse Nuclear Safety Department, dated May 22, 1981.

2. Letter from W. J. Johnson (W) to R. C. Jones (NRC), "Use of 2700°F Acceptance Limit in Non-LOCA Accidents," NS-NRC-89-3466, dated October 19, 1989.

3. Appendix E of the March 23, 1972 Testimony of Westinghouse Electric Corporation, in the matter of the Interim Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors (Docket No. RM50-1).

4. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), WCAP-7907-A, April 1984.

5. Hargrove, H. G., "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

Design Category - Fuel Coolability Criteria

Design Parameter - Clad Ballooning and Flow Blockage

Design Basis - Appendix K to 10CFR50 requires that an accurate model is used in LOCA analyses to predict clad strain, burst strain and the assembly flow blockage.

Acceptance Limit - The models utilize applicable test data in such a way as to properly estimate the pre-rupture clad strain, the rupture (burst) strain at the location of clad rupture and not underestimate the assembly flow blockage, References 1 and 2.

Design Evaluation - The large break LOCA evaluation, References 3, 4, 5, and 6 uses the model described in Reference 1. The small break LOCA evaluation, References 7 and 8, uses a slightly different model which is documented in Reference 2.

Reference(s)

1. Powers, D.A. and Meyer, R.O., "Cladding Swelling and Rupture Models for LOCA Analysis", USNRC Report NUREG-0630, April 1980.

2. Skwarek, R., et al., "Westinghouse Emergency Core Cooling System Small Break, October 1975 Model," WCAP-8970-P-A, January 1979.

3. "Westinghouse ECCS Evaluation Model 1981 Version", WCAP-9220-P-A Revision 1, February 1982.

4. Young, M. Y., et al., "BART-A: A Computer Code for the Best Estimate Analysis of Reflood Transients", WCAP-9561-P-A, March 1984.

5. Besspiata, J. J., et al., "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code", WCAP-10266-P-A, Revision 2, with Addenda, March 1987.

6. Hochreiter, L. E., et al., "Westinghouse Large Break LOCA Best Estimate Methodology," WCAP-10924-P-A Rev. 1, December 1988.

7. Lee, N., et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A, August 1985.

8. Meyer, P. E., "NOTRUMP, A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A, August 1985.

Design Category - Fuel Coolability Criteria

Design Parameter - Violent Expulsion of Fuel (Rod Ejection)

Design Basis - Violent expulsion of fuel material as a result of a Control Rod Ejection Accident will be avoided.

Acceptance Limit - The Westinghouse design limit for the average fuel pellet enthalpy is 200 cal/g (360 Btu/lbm) for irradiated and unirradiated fuel, Reference 1. The NRC acceptance limit is 280 cal/g.

Design Evaluation - For the Control Rod Ejection Accident, fuel pellet enthalpy is calculated using codes and methods which have either been approved by the NRC or accepted by the NRC in specific reviews, References 2, 3, and 4.

Reference(s)

1. Risher, D. H., Jr., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," WCAP-7588, Revision 1-A, January 1975.

2. Risher, D. H., Jr. and R. F. Barry, "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979-P-A, January 1975.

3. Hargrove, H. G., "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

4. Letter from W. J. Johnson (W) to R. C. Jones (NRC), "Use of 2700°F Acceptance Limit in Non-LOCA Accidents," NS-NRC-89-3466, dated October 19, 1989.

Design Category - Fuel Coolability Criteria

Design Parameter - Fuel Assembly Structural Response to Seismic/LOCA Loads

Design Basis - Fuel rod fragmentation will not occur as a direct result of the blowdown load. The 10CFR50.46 temperature and oxidation limits will not be exceeded, and damage will not be so severe as to prevent control rod insertion.

Acceptance Limit - Either the combined loads on the fuel rod spacer grid remain below the allowable grid loads, (control rod insertion will not be interfered with by lateral displacement of the guide tubes) or analysis demonstrates that the grid deformation is not severe enough to prevent control rod insertion and the peak cladding temperature and local oxidation remain below the specified limits in 10CFR50.46, References 1 and 2.

Design Evaluation - The structural adequacy of fuel assemblies is evaluated using the requirements for combined seismic and LOCA load per Appendix A to the NRC SRP 4.2 and the approved analysis methodology, Reference 3 . The analytical procedures given in Reference 3 are used to assess fuel designs.

For plants in which the spacer grid loads are determined to exceed the allowable grid loads, it is demonstrated that both the fuel core coolability and the requirements for control rod insertability for safe shutdown are met. Either best estimate or Appendix K models may be used.

Reference(s)

1. Davidson, S. L. and Iorii, J. A., "Reference Core Report - 17x Optimized Fuel Assembly," WCAP-9500-A, May 1982.

2. Davidson, S. L. and Kramer, W. R. (Eds.), "Reference Core Report VANTAGE 5 Fuel Assembly, WCAP-10444-P-A, September 1985.

3. Davidson, S. L. and Iorii, J. A. (Eds.) "Verification Testing and Analyses of the 17x17 Optimized Fuel Assembly," WCAP-9401-P-A, August 1981.

Design Category - Nuclear Design Criteria

Design Parameter - Shutdown Margin

Design Basis - The core will be subcritical at its most reactive condition throughout reactor life by an amount equal to the minimum shutdown margin specified in the plant's Technical Specifications. In all analyses involving reactor trip, the single highest-worth rod cluster control assembly will be postulated to remain untripped in its full-out position (stuck rod criterion).

When fuel assemblies are in the pressure vessel and the vessel head is not in place, k_{eff} will be maintained at or below the Technical Specification value with all control rods inserted and minimum refueling boron.

Acceptance Limit - The acceptance limit is the minimum shutdown margin as specified in each plant's Technical Specifications for each permissible reactor operating condition (e.g., hot power, hot standby, hot shutdown, and cold shutdown conditions).

Design Evaluation - The methods used for shutdown margin evaluation are given in Reference 1.

Reference

1. Davidson, S.L. (ed.) et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272-P-A, July 1985.

Design Category - Nuclear Design Criteria

Design Parameters - Fuel Storage Subcriticality

Design Basis - The design basis for preventing criticality in the spent fuel storage racks is that, including uncertainties, there is a 95 percent probability at a 95 percent confidence level that the effective multiplication factor (k_{eff}) of the fuel assembly array in the fuel storage racks will be less than 0.95.

Including all calculational uncertainties the multiplication factor, k_{eff}, of the fresh fuel assembly array will be less than 0.95 in the normal dry condition or in the abnormal completely water-flooded condition. Additionally, the k_{eff} will not exceed 0.98 with all but one of the non-storage vault covers in place when optimum moderation (foam, spray, fogging or small droplets) is assumed.

Acceptance Limit - ANSI Standard N210-1976, Reference 1, specifies k_{eff} not to exceed 0.95 in spent fuel storage racks and transfer equipment flooded with pure water. For spent fuel storage application, water is usually present.

The design methodology and evaluation also prevent accidental criticality when fresh fuel assemblies are stored in the dry condition. For this case possible sources of moderation such as those that could arise during fire fighting operations are included in the analysis. The design basis k_{eff} is 0.98 as recommended in ANSI N210-1976.

Design Evaluation - The design method which insures the criticality safety of fuel assemblies outside the reactor uses the AMPX system of codes, References 2 and 3, for cross-section generation and KENO IV, Reference 4, for reactivity determination.

References

1. "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations," ANS-57-2, ANSI N210-1976, April 1976.

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

3. Greene, N. M. et al., "AMPX: A Modular Code System for Generating Coupled Multi-group Neutron-Gamma Libraries from ENDF/B," ORNL/TM-3706, March 1976.

4. Petrie, L. M. and Cross, N. F. "KENO-IV - An Improved Monte Carlo Criticality Program," ORNL-4938, Nov. 1975.

Design Category - Nuclear Design Criteria

Design Parameter - Stability

Design Basis - Any spatial power oscillations within the core will be readily detected and suppressed.

Acceptance Limit - The available control system must be able to dampen the power oscillations.

Design Evaluation - Due to the negative power coefficient of reactivity, PWR cores are inherently stable to oscillations in total power. The stability of the turbine/steam generator/core systems and the reactor control system is such that total core power oscillations are not normally possible. The core is designed so that diametral and azimuthal oscillations due to spatial xenon effects are self-dampening and no operator action or control action is required to suppress them.

Even though axial xenon spatial power oscillations may occur, they are readily controllable. The control bank and excore detectors are provided for control and monitoring of axial power distributions. Assurance that fuel design limits are not exceeded is provided by the reactor protection system and the limiting conditions for operation given in the plant's Technical Specifications.

The evaluation of the stability of the core to xenon-induced power oscillations is performed using NRC approved codes, Reference 1.

Reference(s)

1. Davidson, S. L. and Iorii, J. A. "Reference Core Report - 17x17 Optimized Fuel Assembly," WCAP-9500-A, May 1982.

Design Category- Nuclear Design Criteria

Design Parameter - Reactivity Feedback Coefficients

Design Basis - The fuel temperature coefficient will be negative. The moderator temperature coefficient, if positive, will be limited to values such that the net reactivity feedback characteristics are negative and yield acceptable consequences for limiting postulated transients and accidents, and assure adequate core power level control and maneuvering.

Acceptance Limit - Since the fuel and moderator temperature coefficients change during the life of the core, ranges of coefficients are employed in the transient analysis to determine the response of the plant. Coefficient values as a function of time will be such that all applicable transient and accident limits are met.

Design Evaluation - The methods used for evaluating the reactivity feedback coefficients are given in Reference 1.

Reference(s)

1. Davidson, S.L. (ed.) et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272-P-A, July 1985.

Design Category - Nuclear Design Criteria

Design Parameter - Power Distribution

Design Basis - With at least a 95 percent confidence level:

1. The fuel peak linear heat rate will be limited such that the Final Acceptance Criteria (10CFR50.46) for LOCA are satisfied.

2. Under abnormal conditions, including the maximum overpower condition, the fuel peak linear heat rate will not cause melting.

3. Under abnormal conditions, including the maximum overpower condition, the fuel will not operate with a power distribution that would result in the departure from nucleate boiling (DNB) design basis.

Acceptance Limit - The fuel peak linear heat rate limit, $F_Q(z) \times P_{rel}$ evaluated as a function of core height, shall satisfy the Final Acceptance Criteria Limit (10CFR50.46) for the LOCA, as found in the plant's Technical Specifications.

The centerline temperature at the fuel peak linear heat rate resulting from overpower transients/overpower errors is below that required to produce fuel centerline melting, including allowances for uncertainties, over the fuel lifetime.

The permitted relaxation of $F\Delta H$ with decreasing core power must be reflected in the DNB protection setpoints.

Design Evaluation - The methods used for evaluating power distribution are given in References 1, 2, 3 and 4.

Reference(s)

1. Morita, T., et al., "Power Distribution Control and Load following Procedures," WCAP-8385, September 1974.

2. Davidson, S. L. (ed.) et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272-P-A, July 1985.

3. Miller, R. W., et al., "Relaxation of Constant Axial Offset Control, F_Q Surveillance Technical Specification," WCAP-10216-P-A, August 1982.

4. Fici, J. A., et al., "Design Basis for the Thermal Overpower Delta T and Thermal Overpower Delta-T Trip Functions, WCAP-8745-P-A, September 1986.

Design Category - Nuclear Design Criteria

Design Parameter - Maximum Controlled Reactivity Insertion Rate

Design Basis - The maximum reactivity worth of control rods and the maximum rates of reactivity insertion employing control rods will preclude rupture of the coolant pressure boundary or disruption of the core internals to a degree which would impair core cooling capacity due to a rod withdrawal or ejection accident. The fuel peak linear heat rate and DNBR will not exceed the maximum allowable fuel limits at the overpower conditions.

Acceptance Limit - The DNBR does not fall below its acceptance limit due to an accidental withdrawal of a control bank (or banks).

Design Evaluation - The methods used for evaluating the maximum reactivity insertion rate are given in Reference 1.

Reference(s)

1. Davidson, S. L. (ed.) et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272-P-A, July 1985.

5.0 LEAD TEST ASSEMBLIES APPLICATION

Lead Test Assemblies (LTAs) are inserted in PWR cores to obtain early irradiation experience on new product features subjected to normal operating conditions. Several guidelines associated with the LTA demonstration programs have been accepted and include: (1) the design of the LTAs are mechanically and hydraulically compatible with existing fuel assemblies, (2) the peaking factors meet the Technical Specification limits, (3) NRC approved/accepted safety/design evaluation methodology and codes described in Section 4.0 are used, (4) no specified acceptable fuel design limits (SAFDLs) described in Section 4.0 of the FCEP process are exceeded, and (5) not more than eight (8) LTAs per core are normally inserted.

The insertion of the LTAs in a core is a change in the plant facility as described in the plant's FSAR. All safety evaluations in support of using LTAs are performed in accordance with accepted methodologies defined in Section 4.0. It can be concluded that Technical Specification changes are not required and that unreviewed safety questions do not result due to the insertion of LTAs. Therefore, utilities participating in LTA programs need not obtain prior NRC approval for this change, if the review of the reload core results in the conclusions stated above. See Reference 2.

Reference(s)

1. Davidson, S. L. (Ed.) et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272-P-A, July 1985.

2. 10CFR50.59 Section (a) Code of Federal Regulations, Title 10 - Energy, Chapter 1 - NRC, Part 50.59 - Changes, Tests and Experiments, Section (a).

SECTION 6.0 EVALUATION OF METHODOLOGY CHANGES

Section 4.0 of this report references many codes and methodologies that have been approved by the NRC. Westinghouse periodically updates these codes and methods as additional data and new technologies become available and then submits them to the NRC for review and approval. Once these codes or methods are approved by the NRC, they are incorporated into our design process where applicable. In other words, they will be implemented in the design process on a plant specific basis, provided that the codes and methods approved by the NRC are applicable to the specific plant in question. Also, some fuel changes approved by the NRC reference specific methods that were approved by the NRC at the time the SER was issued. Subsequent to the SER issuance, some of these methods have been upgraded and approved by the NRC. These newer methods, if applicable, will continue to be utilized by Westinghouse in evaluating changes.

In order to make this a "living" document, these codes and/or methods will be used to perform the evaluations described in this document without immediately amending this document. This document will be updated and submitted to the NRC for information, if significant new methodologies have become approved, no more frequently than every three years.

In addition to making fuel system changes which meet the previously stated criteria, there could be a need to make small extensions to methodologies that have already been approved, for example, in the area of DNB. Additionally, there could be a need to extend burnup slightly beyond the licensed limit established in WCAP-10125-P-A. Westinghouse will utilize the approaches described on the following pages to validate extensions in methodology and burnup limits. In the event such extensions are not validated, appropriate documents will be submitted to the NRC for review and approval.

DNB CORRELATION

The fundamental criterion that must be met for core thermal and hydraulic design is the DNB design basis. The DNB design basis is that the probability of the limiting fuel rod not being in DNB be greater than 95 percent at a 95 percent confidence level.

An existing DNB correlation will be valid and will meet the above design basis without reservation provided the new fuel assembly geometry is [.]$^+$ The relevant geometric parameters of the test data are: [ ]$^+$ Typical correlation parameters are [ ]$^+$ Exact correlation parameters are a function of the DNB correlation in question. Specific mixing vane designs encompass [ ]$^+$ The relevant fluid parameters of the test data are [ ]$^+$ $^{+(a,c)}$

If the new geometry is [ ]$^+$ of the test data, Westinghouse will evaluate the geometry [ ]$^+$ $^{+(a,c)}$

If additional test data are used, [ ]$^+$ The new data will then be [ ]$^+$ $^{+(a,c)}$

If the new data [ ]$^+$ it may be treated explicitly as a [ ]$^+$ may be developed. If this step is necessary, it would involve NRC review. $^{+(a,c)}$

The results of the evaluation will either (1) demonstrate the continued applicability of an existing DNB correlation to meet the above design basis [

]$^{+}$ $^{+}$(a,c)

EXTENDED BURNUP

The current licensing limit for burnup of Westinghouse fuel is that the lead rod average burnup will not exceed [ ]+ MWD/MTU. Extension of the lead rod average burnup to a value +(a,c) beyond this limit value will be acceptable provided that the following conditions are met:

1. The radiological consequences of the increased burnup are shown to be acceptable (Criteria specified in WCAP-10125-P-A).

2. The maximum fuel rod average burnup does not exceed [ ]+ MWD/MTU. +(a,c)

3. The design bases and NRC approved Westinghouse design limits given in Section 4.0 will be shown to be satisfied.

4. All fuel rods which exceed the current licensing burnup limit will be shown to operate with steady state rod average power levels that are 10% less than the rated core average linear power during the portion of their exposure period when the current lead rod burnup limit is exceeded.

5. The maximum number of fuel assemblies which contain fuel rods with rod average burnup in excess of the current limit will be less than 20% of the fuel assemblies in the core.

The basis for these acceptance criteria for increased discharge burnup is that the maximum extrapolation beyond the current burnup limit is small (approximately 16%), the number of assemblies with these increased burnup levels in the core is limited, and the allowable duty in these rods will be well below core average during their period of operation at extended burnup.

SECTION 7.0 FUEL PERFORMANCE AND MATERIAL PROPERTIES MODELS

The Westinghouse PAD3.4 fuel performance code, Reference 1, is the basic tool used by Westinghouse to perform fuel rod mechanical design analyses and to generate fuel temperature and rod internal pressure input to safety analyses. The PAD3.4 performance models have been developed on a best estimate basis, with appropriate model uncertainties derived to assure that adequate conservatism is incorporated in all fuel rod design and safety related calculations. Fuel and cladding material properties models, References 2 and 3 (for ZIRLO™), used in Westinghouse design and safety evaluations have also been reviewed by NRC.

The Fuel Criteria Evaluation Process will be applied to adjustments of existing PAD fuel performance and material properties models, based on the analysis of new data. This ability to update PAD models as additional performance data becomes available is important to assure that all fuel performance analysis and design calculations appropriately reflect expected behavior. The revised fuel performance models will be established to produce consistent overall agreement between the model prediction and the updated fuel performance data base. The use of performance and material property model uncertainties, typically defined to bound at least 95% of the model data base, assures that analyses performed with the updated PAD3.4 code will remain appropriately conservative.

Modified fuel performance and material property models will employ the [

]$^{+}$ will not be implemented using the Fuel Criteria Evaluation process. +(a,c)
These model revisions will be submitted for explicit NRC review and approval.

This application of the Fuel Criteria Evaluation Process to fuel performance and materials properties models will permit model revisions, including extensions of the range of applicability of the models, without formal NRC review provided that the following conditions are satisfied:

1) The chang. in the integrated fuel performance code response for

   a) maximum fuel average temperature,

   b) end of life rod internal pressure, and

   c) cladding strain,

   relative to that predicted using the NRC approved PAD3.4 code version for established benchmark cases will not exceed the limits defined in Table 7.1.

2) The change in the predictive capability for each model which is modified, relative to the reference performance model data base for the NRC approved PAD code version, will be within the value specified for each model in Table 7.2. Only those models specified in Table 7.2 will be subject to revision under the Fuel Criteria Evaluation Process.

3) New fuel performance or material property da'a sets to be used in developing a modified performance or material property model will be evaluated relative to the current NRC approved model prior to incorporation into an updated model development data base. Only those data sets which meet the limits specified in Table 7.2 relative to the current appr-ved model will be used in model adjustments under the Fuel Criteria Evaluation Process.

4) New fuel performance models will not be introduced using the Fuel Criteria Evaluation Process. Any new performance phenomena to be modeled in a revised PAD version will be submitted to the NRC for explicit review and approval. Subsequent to NRC approval of a new model, future revisions to that model will be subject to the Fuel Criteria Evaluation Process.

5) The Fuel Criteria Evaluation Process will not be used to extend the range of applicability of any fuel performance or material property model to burnups [ ]+ beyond the target lead rod average licensed burnup limits approved for Westinghouse fuel by the NRC. +(a,c)

6) The Fuel Criteria Evaluation Process will not be used to extend the applicability of fuel performance or material property models to new materials, where new materials are defined as materials which have not explicitly been reviewed and approved for use in reactors by the NRC.

Documentation of the basis for all model adjustments will be maintained, including appropriate statistical tests for goodness of fit and variability. Uncertainties in the models will be addressed and bounding models (95% bounding) will be established using all available applicable data as required to appropriately incorporate the impact of uncertainties in fuel licensing analyses. Westinghouse will transmit to the NRC for information any fuel performance and material property model modifications implemented under the Fuel Criteria Evaluation Process within six months of implementation.

Benchmark cases have been defined for evaluating the change in the integrated fuel performance code response per the limits identified in Table 7.1. These benchmark cases include [

]$^{+}$ $^{+}$(a,c)

[

$]^{+}$ $^{+}$(a,c)

Should the fission gas release data for higher gadolinia concentrations indicate that a model adjustment is required, if the adjusted model satisfies the conditions specific in Tables 7.1 and 7.2, the adjusted model will be documented and implemented as part of the Fuel Criteria Evaluation Process. Should the revised models not satisfy the conditions in Tables 7.1 and 7.2, then an explicit NRC review of the proposed model change would be required prior to implementation.

All fuel rod designs which employ gadolinia concentrations [$]^{+}$ will be shown to satisfy all of the criteria established in Section 4 of the Fuel Criteria Evaluation Process. $^{+}$(a,c)

References:

1. Weiner, R. A., et al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851-P-A (Proprietary) and WCAP-11873-A (Non-Proprietary), August 1988.

2. Kuchirka, P. J., "Properties of Fuel and Core Component Material," WCAP-9179, Revision 1, July 1978.

3. Davidson, S. L. and Nuhfer, D. L., Ed., "VANTAGE+ Fuel Assembly Reference Core Report," WCAP-12610, June 1990.

4. NRC Memorandum, C. E. Rossi (NRC) to E. P. Rahe, Jr. (W), "Acceptance for Referencing of Licensing Topical Report WCAP-8720 Addenda 3 'Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations/Application for BWR Fuel Analysis,'" September 15, 1986.

5. Ewing, J. H. and Smith, W. L., "Improved Analytical Models used in Westinghouse Fuel Rod Design Computations/Application for BWR Fuel Analysis," WCAP-8720, Addendum 3, January 1983.

6. NS-EPR-2835, "Response to Request for Additional Information on WCAP-8720, Addendum 3," November 4, 1983.

7. NS-NRC-86-3145, "Response to NRC Gadolinia Questions on Westinghouse Topical Report, WCAP-8720, Addenda 3," June 30, 1986.

Table 7.1
Limits for Allowable Change in the Integrated
Fuel Performance Model Response to Model Updates

+(a,c)

Table 7.2
Limits for Allowable Change in the Individual Performance and Material Property Models
Relative to the Reference Model Data Base

+(a,b,c)

SECTION E

APR 2 9 1992
E. H. NOVENDSTERN

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

April 24, 1992

Mr. E. H. Novendstern, Manager
T&H Design and Fuel Licensing Department
Westinghouse Electric Corporation
Box 3912
Pittsburgh, PA 15230

Dear Mr. Novendstern:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION ON WCAP-12488, "WESTINGHOUSE FUEL CRITERIA EVALUATION PROCESS" (TAC NO. M77257)

We are currently reviewing the Westinghouse Topical Report WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process." The initial review reveals the need for additional information requested in the enclosure. You are requested to respond to these questions as expeditiously as possible in order for us to complete the review. Should you have any questions regarding this request for information, please contact Mr. S.L. Wu of my staff at (301)504-3284.

Sincerely,

Robert C. Jones, Chief
Reactor Systems Branch
Division of Systems Technology

Enclosure:
Request for Information

QUESTIONS ON WESTINGHOUSE FUEL CRITERIA EVALUATION PROCESS REPORT WCAP-12488

This review is of Westinghouse's (W) proposal in the subject topical report to establish criteria that will allow W to make relatively minor fuel design changes without U.S. Nuclear Regulatory Commission (NRC) review and approval. These criteria would also establish when such proposed changes will require NRC review and approval. The review of the submitted topical report WCAP-12488 has raised questions that can be classified into the following three categories:

1. Clarifying specific design bases, limits, and evaluation methods currently used by W in their licensing analyses.
2. Clarifying the criteria that W will apply to new materials, designs, or evaluation methods to determine whether or not NRC review and approval will be required.
3. Clarifying how the requested extension in fuel rod-average burnup will impact the material/design basis limits and evaluation methods used to support the licensing analyses. Also those data that W have to support this extension in fuel burnup.

CATEGORY 1

1. The section entitled "Fuel Assembly Structural Response to Seismic/LOCA Loads" (page 34) states that either best estimate or Appendix K models may be used when allowable spacer grid loads are exceeded to demonstrate that requirements for core coolability and control rod insertability for safe shutdown are met. Have the best estimate models been reviewed and approved by NRC? Please elaborate and be more specific on how the approved best estimate models will be used.

CATEGORY 2

1. Additional criteria are needed to define those conditions when a new design will or will not be submitted to NRC for review and approval. Please provide these additional criteria. These additional criteria should also address the introduction of new materials in a design. Of particular concern to NRC are those new designs/materials that significantly impact a) the results of design basis accidents, b) the applicability of NRC approved evaluation models, and/or c) the applicability of operating/design limits [with the exception of DNBR limits as discussed in the subject report], compared to previously approved fuel designs. Please provide examples of design/material changes for which an NRC review would and would not be required based on these proposed criteria.

2. In Section 3.1 of the submittal it is stated that utilities may supply portions of the calculations, using codes similar to those used by W, to support a specific design application. This implies that utility models/codes and calculational methods will be used in some instances. What criteria will be used to determine that the utility codes and analyses are applicable and acceptable to the specific design? Who is responsible for making this determination and how is it determined that the utility has the expertise for making these analyses?

3. The following questions are related to thermal hydraulic core analyses:

   (a) When an existing CHF correlation and DNBR limit, i.e., one that was developed for an existing design, is being proposed for application to a new design, what criteria or tests will be applied to the geometric and correlation parameters to determine that the new assembly is similar to or bracketed by "the test data and correlation parameters"? Conversely, what criteria or tests will be used to determine that thermal hydraulic and CHF tests are needed for a new design? Please provide an example where a new design is determined to be bracketed by existing test data and correlation parameters and a second example where the new design is not bracketed by existing DNB data and correlation parameters.

   (b) If new thermal hydraulic tests are performed, what statistical tests will be used to determine if the new DNB data for the new design are from the same population as the existing data base? Please provide an example application of the statistical methods that will be used to determine if the new data base is from the same population as the existing data base. How will the range of test parameters be chosen for the experimental data from the new design and how will it be determined that the number of test data are satisfactory? Also, please provide an example of the statistical methods that will be used to determine a new correlations' 95/95 limit with the "pooled" data base.

4. Three alternative methods are provided in the topical report for evaluating fretting wear depth. Please discuss the criteria that will be applied to determine which of the three alternative methods will be used for evaluating fretting wear. For example, it is anticipated that the second method will be acceptable only if the design being considered and the results of out-of-reactor tests meet particular criteria in relation to previous designs and their out-of-reactor results. Please provide examples. The third method of calculating fretting wear depth includes the use of semi-empirical models. Please provide a description and justification (including comparisons of the empirical model to actual in-reactor wear data up to the burnup limit requested) of why and when these semi-empirical models are justified. Also provide an example of when this third method would be used.

CATEGORY 3

1. The recently observed reduction in Zircaloy-4 cladding ductility to values between 0.03 and 0.11% at local burnup levels between 55 and 63 MWd/kgM has raised the concern that mechanical limits imposed to prevent fuel failures during normal and anticipated operational occurrences (AOOs) are non-conservative at and, particularly, above these burnup levels. The cladding mechanical limits of particular concern are a) the uniform elastic plus plastic strain limit of 1.0%, b) the 0.2% irradiated yield strength, and c) the O'Donnel and Langer fatigue curves for irradiated cladding. In addition, no ramp power testing has been performed on prototypic commercial fuel rods to determine failure thresholds at burnup levels greater than ~48 GWd/MTM. Please provide data and analyses that demonstrate that these mechanical design limits remain conservative up to the burnup level requested for Zr-4 cladding or provide alternative mechanical limits that can be defended as being conservative at these higher burnup levels. Also, please provide bounding analyses that demonstrate how much margin exists between calculated values of strain and strain fatigue and their respective design limits up to the requested extension in burnup. How much have these margins decreased as a result of this extension in burnup?

2. It is not apparent that the cladding temperature limits proposed by W will exclude excessive fuel cladding oxidation at the extended burnup level requested. The NRC Standard Review Plan (SRP) recommends that the effects of cladding oxidation be included in thermal and mechanical analyses. Please discuss how W will include the effects of cladding oxidation in their thermal analyses. In the past, W has reduced cladding thickness by a proprietary amount to account for cladding imperfections, fretting wear and waterside corrosion. There is a concern that some W plants with high primary coolant temperatures and elevated lithium may exceed the W reduction in cladding thickness for mechanical analyses and the oxidation thickness used for thermal analyses at the extended burnup levels requested. This is of particular concern since waterside corrosion has been shown to accelerate at extended burnup levels. Please provide the relevant in-reactor waterside corrosion data and how they are applied to thermal and mechanical analyses that demonstrate that W conservatively accounts for the effects of Zircaloy-4 waterside corrosion at the burnup limit requested. Also, because cladding waterside corrosion is reactor specific, how does W intend to evaluate this phenomenon for those plants with a past history of high Zircaloy corrosion either due to poor controls on water chemistry, high coolant temperatures or elevated lithium levels?

3. Does W have more recent rod and assembly growth data up to the burnup limit requested? If so, please demonstrate that based on this new data the axial growth models will remain conservative up to the burnup requested.

SECTION F

Westinghouse Electric Corporation
Energy Systems
Box 355
Pittsburgh Pennsylvania 15230-0355

June 8, 1992
ET-NRC-92-3702

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Attention: R. C. Jones, Reactor Systems Branch Chief, Division of Engineering and System Technology

Subject: Responses to Request for Additional Information on WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process" [Proprietary]

Reference: (1) Letter from R. C. Jones (NRC) to E. H. Novendstern (Westinghouse), Request for Additional Information on WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process," April 24, 1992.

Dear Mr. Jones:

Enclosed are six (6) copies of the Westinghouse responses [Proprietary] to your request for additional information, Reference 1, on WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process." Your timely review of these responses is requested in order to complete the review of the topical.

This submittal contains Westinghouse proprietary information of trade secrets, commercial or financial information which we consider privileged or confidential pursuant to 10CFR9.5(4). Therefore, it is requested that the Westinghouse proprietary information attached hereto be handled on a confidential basis and be withheld from public disclosure.

This material is for your internal use only and may be used solely for the purpose for which it is submitted. It should not be otherwise used, disclosed, duplicated, or disseminated, in whole or in part, to any other person or organization outside the Office of Nuclear Reactor Regulation without the express written approval of Westinghouse.

Correspondence with respect to any Application for Withholding should reference AW-92-322 and should be addressed to N. J. Liparulo, Manager of Regulatory and Legislative Affairs, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

Nicholas J. Liparulo

Nicholas J. Liparulo, Manager
Nuclear Safety and Regulatory Activities

cc: L. Barnett - NRC (MS 12H5)
L. E. Phillips - NRC (MS 8E23)
S. L. Wu - NRC (MS 8E23)

Westinghouse Electric Corporation
Energy Systems
Box 355
Pittsburgh Pennsylvania 15230-0355

June 8, 1992
AW-92-322

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Attention: Mr. R. C. Jones, Reactor Systems Branch Chief, Division of Engineering and System Technology

Reference: Letter from N. J. Liparulo (Westinghouse) to R. C. Jones (NRC), ET-NRC-92-3702, dated June 8, 1992

Subject: Responses to Request for Additional Information on WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process," [Proprietary]

Dear Mr. Jones:

The above referenced letter contains information proprietary to the Westinghouse Electric Corporation.

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

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

Correspondence with respect to the proprietary aspects of this submittal should reference AW-92-322 and should be addressed to the undersigned.

Very truly yours,

Nicholas J. Liparulo

Nicholas J. Liparulo, Manager
Nuclear Safety and Regulatory Activities

cc: M. P. Siemien, Esq.
Office of the General Counsel, NRC

Westinghouse Responses To NRC Questions On The Westinghouse Fuel Criteria Evaluation Process (FCEP) WCAP-12488

CATEGORY 1

Question 1. *The section entitled "Fuel Assembly Structural Response to Seismic/LOCA Loads" (page 34) states that either Best Estimate or Appendix K models may be used when allowable spacer grid loads are exceeded to demonstrate that requirements for core coolability and control rod insertibility for safe shutdown are met. Have the best estimate models been reviewed and approved by the NRC? Please elaborate and be more specific on how the approved best estimate models will be used.*

RESPONSE QUESTION 1

The SECY-83-472 best estimate approach has been approved by the NRC. This approach is not a true best estimate approach but was developed to address concerns for Westinghouse plants which have upper plenum injection. These plants are primarily of the Westinghouse two (2) loop design, which are much older than the current generation of Westinghouse plants and many are not required to include seismic loads in the structural analysis of the fuel. Therefore, grid deformation is not an issue for these plants. Further, Westinghouse does not currently plan to extend the SECY-83-472 best estimate approach to other designs. Therefore, the issue for modeling of deformed grids does not apply.

The new Westinghouse Best Estimate models which are being developed under the provision of 10CFR50.46(a)(1)(i), as amended in October 1988, are in development and have not received NRC approval. The FCEP approach would only account for best estimate effects for plants licensed with NRC approved best estimate LOCA models. Since these models are still under NRC review and application to operating Westinghouse PWRs -has not yet begun, the issue of how to model grid deformation has not been addressed. When grid deformation must be addressed for application of best estimate technology, Westinghouse would develop the necessary models at that time and submit the grid deformation/core coolability methodology to the NRC as part of that plant's licensing application.

CATEGORY 2

Question 1. *Additional criteria are needed to define those conditions when a new design will or will not be submitted to NRC for review and approval Please provide these additional criteria. These additional criteria should also address the introduction of new materials in a design. Of particular concern to NRC are those new designs/materials that significantly impact a) the results of design basis accidents, b) the applicability of NRC approved evaluation models, and/or c) the applicability of operating/design limits [with the exception of DNBR limits as discussed in the subject report], compared to previously approved fuel designs. Please provide examples of design material changes for which an NRC review would and would not be required based on these proposed criteria.*

RESPONSE QUESTION 1

When a new design or material change is introduced, it is possible that the criteria may not comply with those contained in WCAP-12488. Should this happen, that aspect will be submitted to the NRC for separate review. Subsequent to the NRC approval, the new criteria will be used without future submittals to the NRC.

The application of the Fuel Criteria Evaluation Process is intended for fuel mechanical design changes which are extensions/minor changes to existing fuel assembly/component designs previously reviewed and accepted by the NRC provided that all the established criteria in WCAP-12488 are satisfied. An example of this is the VANTAGE 5H fuel assembly. In this example, the criteria discussed in WCAP-12488 were addressed to insure the acceptability of this design change. It was shown using NRC approved methods/models that compared to the previously approved advanced fuel assembly designs the VANTAGE 5H design change did not impact the design basis incidents or operating/design limits.

The application of the Fuel Criteria Evaluation Process is intended for designs which employ materials and materials properties previously reviewed and accepted by the NRC. Designs which employ materials addressed in WCAP-9179 do not require additional materials specific review by the NRC, provided that all of the established criteria in WCAP-12488 are satisfied. Examples of materials for which specific licensing would not be required under this process include improved Zircaloy-4 cladding and low cobalt bottom nozzles. In both of these examples, the materials which are being used have compositions which lie within the ranges specified for those materials in WCAP-9179. An example of a materials change that would require explicit NRC review is the implementation of a new cladding alloy, such as ZIRLO™ cladding. Although ZIRLO™ is similar to Zircaloy, the criteria of acceptance (10CFR50.46 and 10CFR50 Appendix K) are specifically identified as appropriate for Zircaloy clad fuel. Therefore, an NRC review was required and exemptions were obtained to allow application of those criteria to ZIRLO™ clad fuel. However, once a generic review of ZIRLO™ cladding material is completed by the NRC, new fuel designs which employ ZIRLO™ cladding would not require specific NRC review, provided that all of the established criteria in WCAP-12488 are satisfied.

Question 2 *In Section 3.1 of the submittal it is stated that utilities may supply portions of the calculations, using codes similar to those used by Westinghouse, to support a specific design application. This implies that utility models/codes and calculational methodology will be used in some instances. What criteria will be used to determine that the utility codes and analyses are applicable and acceptable to the specific design Who is responsible for making this determination and how is it determined that the utility has the expertise for making these analyses?*

RESPONSE QUESTION 2

The criterion used by Westinghouse to determine if utility codes and analyses are applicable and acceptable to a specific design is that utilities who provide analyses of record must have them explicitly approved by the NRC.

This would be done consistent with established precedent, such as the licensing of Northeast Utilities ("Physics Methodology for PAR Reload Design," NUSCO-152, August 1986) for use of Westinghouse Nuclear Design Technology, or other such arrangements established in the future between Westinghouse and the NRC.

Question 3. *The following questions are related to thermal-hydraulic core analyses:*

(a) *"When an existing CHF correlation and DNBR limit, i.e.., one that was developed for an existing design is being proposed for application to a new design, what criteria or tests will be applied to the geometric and correlation parameters to determine that the new assembly is similar to or bracketed by "the test data and correlation parameters"? Conversely, what criteria or tests will be used to determine that thermal-hydraulic and CHF tests are needed for a new design? Please provide an example where a new design is determined to be bracketed by existing test data and correlation parameters and a second example where the new design is not bracketed by existing DNB data and correlation parameters.*

(b) *If new thermal hydraulic tests are performed, what sta tical tests will be used to determine if the new DNB data for the new design are from the same population as the existing database? Please provide an example application of the statistical methods that will be used to determine if the new database is from the same population as the existing database. How will the range of test parameters be chosen for the experimental data from the new design and how will it be determined that the number of test data are satisfactory? Also, please provide an example of the statistical methods that will be used to determine a new correlations' 95/95 limit with the "pooled" database.*

RESPONSE QUESTION 3(a)

When a new fuel assembly design is developed, the geometry is evaluated consistent with the NRC guidelines in the Safety Evaluation Report on the WRB-1 CHF correlation. The geometric parameters critiqued are the fuel rod diameter, rod pitch, heated length, gridded to ungridded cell flow areas, grid spacing, and [ ]$^{+}$ [$^{+(a,c)}$

]$^{+}$ Geometric correlation parameters $^{+(a,c)}$ are a function of the above geometric parameters and thus are not independent. Each geometric parameter should be bounded by the range for grids that have been previously DNB tested and directly support the correlation in question.

If any one of the geometric parameters does not fall within the respective range, justification must be developed to support the new design. This justification [

]+ CHF tests would be required, [+(a,c)

]+ +(a,c)

An example of where a new design was determined to be bracketed by the test data and correlation parameters is the 17x17 VANTAGE 5 Hybrid design. As discussed in Section 4.2.2 of Addendum 2 of WCAP-10444-P-A, this design meets the above mentioned criteria.

At that time, an example of where a new design was not bracketed by the test data or correlation parameters is the 17x17 OFA design. Further elaboration is provided as a response to part (b) of this question.

<u>RESPONSE QUESTION 3(b)</u>

The statistical tests performed would [

] + +(a,b,c)

The 17x17 OFA example is documented in WCAP-9401-P-A.

The test points [

]+ can be made. As the data +(a,b,c)

[

]+ +(a,b,c)

The methods used to determine the pooled database limit DNBR are [

]+ +(a,b,c)

Question 4.

Three alternative methods are provided in the topical report for evaluating fretting wear depth. Please discuss the criteria that will be applied to determine which of the three alternative methods will be used for evaluating fretting wear. For example, it is anticipated that the second method will be acceptable only if the design being considered and the results of out-of-reactor tests meet particular criteria in relation to previous designs and their out-of-reactor results. Please provide examples. The third method of calculating fretting wear depth includes the use of semi-empirical models. Please provide a description and justification (including comparisons of the empirical model to actual in-reactor wear data up to the burnup limit requested) of why and when these semi-empirical models are justified. Also provide an example of when this third method would be used.

RESPONSE QUESTION 4

Past experience with both in-reactor operation and out-of-reactor tests indicate that the most important considerations for successful fretting wear performance are the [ ]$^+$ grid forces. The [ ]$^+$ grid forces must exceed the fretting threshold force at end-of-life (EOL) that have been determined from in-reactor experience and out-of-reactor tests. The EOL forces are predicted based on in-pile creep and relaxation data. $^+$(a,c) $^+$(a,c)

Other significant considerations are the hydraulic conditions (e.g. crossflow generated by pressure drop mismatches) and grid rod support conditions (e.g distance between support points, spring stiffness).

Regarding the determination of the appropriate method for assuring fretting wear performance, three alternative methods were identified in WCAP-12488:

[ ]$^+$ $^+$(a,c)

Use of method #3 is dependent on appropriate [ ]$^+$ Changes in grid design and/or materials can affect [ ]$^+$ In the mid-1970's models such as reported in WCAP-8646 were developed for early Inconel grid designs. More recently, however, these types of models [ ]$^+$ fretting wear evaluation. Where new evaluations [ ]$^+$ $^+$(a,c) $^+$(a,c) $^+$(a,c) $^+$(a,c)

The need for [ ]$^+$ is dependent on how the key considerations identified earlier are affected. For example, if the [ ]$^+$ grid forces are within the fretting experience limits and the material properties are known, then any other changes (e.g. hydraulic conditions or support conditions) outside current experience bounds [ ]$^+$ For example, if increased pressure drop differences [ ]$^+$ Similarly, [ ]$^+$ An example of a design that utilized [ ]$^+$ was VANTAGE 5H (WCAP-10444, Addendum 2A). $^+$(a,c) $^+$(a,c) $^+$(a,c) $^+$(a,c) $^+$(a,c) $^+$(a,c)

[ ]$^+$ $^+$(a,b,c)

[

]+ Examples include Optimized Fuel (OFA) (WCAPs 9500/9400) and +(a,c)
VANTAGE 5 (WCAP-10444).

CATEGORY 3

Question 1. *The recently observed reduction in Zircaloy-4 cladding ductility to values between 0.03 and 0. 11 % at local burnup levels between 55 and 63 MWdlkgM has raised the concern the mechanical limits imposed to prevent fuel failures during normal and anticipated operational occurrences (AOOS) are non-conservative at and, particularly, above these burnup levels. The cladding mechanical limits of particular concern are a) the uniform elastic plus plastic strain limit of 1.0%, b) the 0.2% irradiated yield strength and c) the O'Donnel and Langer fatigue curves for irradiated cladding. In addition, no ramp power testing has been performed on prototypic commercial fuel rods to determine failure thresholds at burnup levels greater than ~ 48 GWd/MTM. Please provide data and analyses that demonstrate that these mechanical design limits remain conservative up to the burnup level requested for Zr-4 cladding or provide alternative mechanical limits that can be defended as being conservative at these higher burnup levels. Also, please provide bounding analyses that demonstrate how much margin exists between calculated values of strain and strain fatigue and their respective design limits up to the requested extension in burnup. How much have these margins decreased as a result of this extension in burnup?*

Question 2. *It is not apparent that the cladding temperature limits proposed by Westinghouse will exclude excessive fuel cladding oxidation at the extended burnup level requested. The NRC Standard Review Plan (SRP) recommends that the effects of cladding oxidation be included in thermal and mechanical analyses. Please discuss how Westinghouse will include the effects of cladding oxidation in their thermal analyses. In the past, Westinghouse has reduced cladding thickness by a proprietary amount to account for cladding imperfections, fretting wear and waterside corrosion. There is a concern that some Westinghouse plants with high primary coolant temperatures and elevated lithium may exceed the Westinghouse reduction in cladding thickness for mechanical analyses and the oxidation thickness used for thermal analyses at the extended burnup levels requested. This is of particular concern since waterside corrosion has been shown to accelerate at extended burnup levels. Please provide the relevant in-reactor waterside corrosion data and how they are applied to thermal and mechanical analyses that demonstrate that Westinghouse conservatively accounts for the effects of Zircaloy-4 waterside corrosion at the burnup limit requested Also, because cladding waterside corrosion is reactor specific, how does Westinghouse intend to evaluate this phenomenon for those plants with a past history of high Zircaloy corrosion either due to poor controls on water chemistry, high coolant temperatures or elevated lithium levels?*

Question 3. *Does Westinghouse have more recent rod and assembly growth data up to the burnup limit requested? If so, please demonstrate that based on this new data the axial growth models will remain conservative up to the burnup requested.*

RESPONSE QUESTIONS 1,2,3

Since the time that WCAP-12488 was prepared for submittal to the NRC, Westinghouse has completed the development of the VANTAGE+ fuel design, which includes the use of ZIRLO^{TM} material for the fuel rod cladding and fuel assembly guide thimble tubes. VANTAGE+ fuel is explicitly designed for high burnup applications beyond the lead rod target burnup established for Westinghouse fuel in WCAP-10125-P-A. The VANTAGE+ fuel design, including increased fuel discharge burnup, has been submitted for NRC review in WCAP-12610. At this time, it is anticipated that all fuel burnup extensions beyond the target level established in WCAP-10125-P-A will employ the VANTAGE+ design. Therefore, the Extended Burnup section in Section 6.0 of WCAP-12488 is no longer relevant, since licensing issues associated with extended burnup are being addressed as part of the NRC review of the VANTAGE+ design in WCAP-12610. Accordingly, no responses will be provided to Category 3, Questions 1, 2 and 3. Should it become desirable in the future to increase the discharge burnup for current fuel designs beyond the established target limit, a burnup specific report will be submitted to the NRC for review and approval.

SECTION G

Westinghouse
Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

July 17, 1992
ET-NRC-92-3723

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Attention: R. C. Jones, Reactor Systems Branch Chief, Division of Engineering and System Technology

Subject: Supplement to Additional Information on WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process" [Proprietary]

Reference: (1) Letter from R. C. Jones (NRC) to E. H. Novendstern (Westinghouse), Request for Additional Information on WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process," April 24, 1992.

(2) Letter from N. J. Liparulo (Westinghouse) to R. C. Jones (NRC), "Responses to Request for Additional Information on WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process, June 8, 1992" [Proprietary}

Dear Mr. Jones:

In response to your request for additional information, Reference 1, on WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process (FCEP), Westinghouse transmitted Reference 2 for your review and approval. In Reference 2, a response to Category 2 question number 1 provided the additional criteria needed to define those conditions when a new design will or will not be submitted to the NRC for review, and specifically addressed the introduction of new materials in a design. Subsequent to the Reference 2 submittal, discussions were held with the Reactor Systems Branch staff on how to address changing the range of applicability of materials models or correlations. It was suggested that Westinghouse submit supplemental information to be incorporated into the FCEP topical. The attached text (6 copies) is submitted for your review and approval and will be incorporated as a new section in WCAP-12488 entitled, "Section 7.0 - Fuel Performance and Material Properties Models." Your timely review of this supplemental information is requested in order to complete the approval of this topical.

This submittal contains Westinghouse proprietary information of trade secrets, commercial or financial information which we consider privileged or confidential pursuant to 10CFR9.5(4). Therefore, it is requested that the Westinghouse proprietary information attached hereto be handled on a confidential basis and be withheld from public disclosure.

This material is for your internal use only and may be used solely for the purpose for which it is submitted. It should not be otherwise used, disclosed, duplicated, or disseminated, in whole or in part, to any other person or organization outside the Office of Nuclear Reactor Regulation without the express written approval of Westinghouse.

Correspondence with respect to any Application for Withholding should reference AW-92-337 and should be addressed to N. J. Liparulo, Manager of Regulatory and Legislative Affairs, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

Nicholas J. Liparulo

Nicholas J. Liparulo, Manager
Nuclear Safety and Regulatory Activities

cc: L. Barnett - NRC (MS 12H5)
L. E. Phillips - NRC (MS 8E23)
S. L. Wu - NRC (MS 8E23)

Westinghouse Electric Corporation
Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

July 17, 1992
AW-92-337

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Attention: Mr. R. C. Jones, Reactor Systems Branch Chief, Division of Engineering and System Technology

Reference: Letter from N. J. Liparulo (Westinghouse) to R. C. Jones (NRC), ET-NRC-92-3723, dated July 17, 1992

Subject: Supplement to Additional Information on WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process," [Proprietary]

Dear Mr. Jones:

The above referenced letter contains information proprietary to the Westinghouse Electric Corporation.

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

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

Correspondence with respect to the proprietary aspects of this submittal should reference AW-92-337 and should be addressed to the undersigned.

Very truly yours,

Nicholas J. Liparulo

Nicholas J. Liparulo, Manager
Nuclear Safety and Regulatory Activities

cc: M. P. Siemien, Esq.
Office of the General Counsel, NRC

SECTION 7.0 FUEL PERFORMANCE AND MATERIAL PROPERTIES MODELS

The Fuel Criteria Evaluation Process may also be applied to adjustments of existing fuel performance and material properties models, based on the analysis of significant new data. The basic equation forms currently reviewed and approved by the NRC will not be modified. Based upon the results of the data evaluation, Westinghouse will either (1) demonstrate the continued applicability of the existing correlation(s), or (2) utilize the previously approved correlations with modified coefficients which produce an improved correlation relative to the total data base. In the event that the correlation form needs to be changed, Westinghouse will submit to the NRC appropriate documentation to substantiate the proposed change for review and approval.

The ranges of applicability of existing models may be extended as additional data becomes available beyond the current range limits. For example, material properties and fuel performance models for Gd_2O_3-UO_2 fuel are currently approved for gadolinia concentrations of up to 6.0% by weight. Additional fuel performance data has become available for increased gadolinia concentrations which may be used to validate existing models for these higher concentrations. Under the Fuel Criteria Evaluation Process, application of existing models to higher gadolinia concentrations consistent with the range of available supporting data could be implemented without explicit NRC review, provided that all of the established criteria in WCAP-12488 are shown to be satisfied. Appropriate documentation of the additional supporting data and associated data analyses will be maintained.

Similarly, as additional fuel rod growth data becomes available for ZIRLO™ cladding, analyses of these data may demonstrate that improvement in the ZIRLO™ rod growth predictions can be obtained by adjustment of the coefficients used in the fuel rod growth equation. It is intended that these coefficient adjustments will be made as part of the Fuel Criteria Evaluation Process, without prior explicit NRC review, provided that all of the established criteria in WCAP-12488 are shown to be satisfied. Documentation of the basis for all model adjustments will be maintained, including appropriate statistical tests for goodness of fit and analysis of variance. Uncertainties in the models will be addressed and bounding models (95% bounding) will be established using all available applicable data as required to appropriately incorporate the impact of uncertainties in fuel performance and design analyses.

(to be added to WCAP-12488)

SECTION H

FAL-92-731

From: Commercial Nuclear Fuel Division
WIN: 284-2104
Date: October 27, 1992
Subj: Telecon on FCEP Topical

To:

Westinghouse*	NRC*	PNL*
E. H. Novendstern	Shih-Liang Wu	C. Beyer
W. Leech		J. Kuda
H. Chelemer		
P. Kersting		
R. Knott - Cola		
G. B. Sieradzki - Cola		

cc: A. L. Casadei
R. S. Miller
T. Bartman - Cola

* telecon participant

As a result of a telecon with the NRC/PNL on October 26th, the following are NRC requests for additional information to close out the review on our FCEP topical, WCAP-12488:

1. Question 3(b). A simple statement stating that the statistical tests performed on WCAP-9401 DNB test data using repeatability parameters, analysis of means and comparison of variances in the responses to NRC questions is not acceptable. It must be demonstrated that the new data is significantly significant. The new database has to be spelled out. Our additional information has to include a data requirement and define what it is and spell out the statistical test requirements for new design changes, i.e., explain in some detail repeatability parameters, analysis of means and comparison of variances in the treatment of test data. This additional information will be used to revise the response to Question 3(b). (Action: W. Leech)

2. Question 3(b). Associated with the additional information above, a procedure needs to be in place to check out the normality of the test data. A procedure to define normality and that the data fit is acceptable is required. Include references to Reg Guide 5.22 and ANSI-15.15 in the response. This additional information will be used to revise the response to Question 3(b). (Action: W. Leech)

3. Section 7. Additional information is required to augment the Section 7 Fuel Performance and Material Properties Models issued to the NRC. For every fuel model change the appropriate acceptance test needs to be included. The NRC suggested that we follow the GE SER on this subject.

1. Predict the behavior of design changes for any model changes.

2. Perform some check on the individual model capabilities; any deviation has to be within some defined parameters.

GE used the fuel centerline temperature for their design parameter. We may want to use, e.g. fission gas release, clad fatigue, fuel rod growth, etc.

The shortfall from the above is that it could result in a change to our performance code such as LOCA. A set of key fuel rod performance design parameters are required to show that the basic performance of the updated code is acceptable. (Action: P. Kersting)

S. L. Davidson

S. L. Davidson
Fuel Analysis & Licensing

/ssh

SECTION I

Westinghouse
Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

February 8, 1993
ET-NRC-93-3819

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Attention: R. C. Jones, Reactor Systems Branch Chief, Division of Engineering and System Technology

Subject: Final Responses to Additional Information on WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process" [Proprietary]

Reference: (1) Letter from N. J. Liparulo (Westinghouse) to R. C. Jones (NRC), ET-NRC-92-3702 "Responses to Request for Additional Information on WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process, June 8, 1992" [Proprietary}

(2) Letter from N. J. Liparulo (Westinghouse) to R. C. Jones (NRC) ET-NRC-92-3723, "Supplement to Additional Information on WCAP-12488, Westinghouse Fuel Criteria Evaluation Process," July 17, 1992 [Proprietary]

Dear Mr. Jones:

Enclosed are fifteen (15) copies of the revised text to Question 3(b) (DNB Test Data) and for the Section 7.0 supplement (Fuel Performance and Material Properties Models) transmitted previously in References (1) and (2) respectively. This text has been revised after discussions with members of your staff and the technical reviewer. This revised text completes all outstanding items concerning the NRC review of WCAP-12488.

This submittal contains Westinghouse proprietary information of trade secrets, commercial or financial information which we consider privileged or confidential pursuant to 10CFR9.5(4). Therefore, it is requested that the Westinghouse proprietary information attached hereto be handled on a confidential basis and be withheld from public disclosure.

This material is for your internal use only and may be used solely for the purpose for which it is submitted. It should not be otherwise used, disclosed, duplicated, or disseminated, in whole or in part, to any other person or organization outside the Office of Nuclear Reactor Regulation without the express prior written approval of Westinghouse.

Correspondence with respect to any Application for Withholding should reference AW-93-411 and should be addressed to N. J. Liparulo, Manager of Regulatory and Legislative Affairs, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

Peter J. Morris / for

Nicholas J. Liparulo, Manager
Nuclear Safety and Regulatory Activities

cc: L. Barnett - NRC (MS [illegible]45)
L. E. Phillips - NRC ([illegible]23)
S. L. Wu - NRC (MS [illegible]

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

February 8, 1993
AW-93-411

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Attention: Mr. R. C. Jones, Reactor Systems Branch Chief, Division of Engineering and System Technology

Reference: Letter from N. J. Liparulo (Westinghouse) to R. C. Jones (NRC), ET-NRC-93-3819, dated February 8, 1993

Subject: Final Responses for Additional Information on WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process [Proprietary]

Dear Mr. Jones:

The above referenced letter contains information proprietary to the Westinghouse Electric Corporation.

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

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

Correspondence with respect to the proprietary aspects of this submittal should reference AW-93-411 and should be addressed to the undersigned.

Very truly yours,

Peter J. Morris/for

Nicholas J. Liparulo, Manager
Nuclear Safety and Regulatory Activities

cc: M. P. Siemien, Esq.
Office of the General Counsel, NRC

Revision to Response to Question 3(b) on WCAP-12488

Question 3(b) *If new thermal hydraulic tests are performed, what statistical tests will be used to determine if the new DNB data for the new design are from the same population as the existing database? Please provide an example application of the statistical methods that will be used to determine if the new database is from the same population as the existing database. How will the range of test parameters be chosen for the experimental data from the new design and how will it be determined that the number of test data are satisfactory? Also, please provide an example of the statistical methods that will be used to determine a new correlations' 95/95 limit with the "pooled" database.*

Response

The statistical tests performed by Westinghouse [ ]+ in WCAP-9401 using standard statistical tests to [ ]+ The procedural steps are: +(a,b,c) +(a,b,c)

[ ]+ +(a,b,c)

The ratio of measured to predicted critical heat fluxes, $\overline{M/P}$, are subjected to the three statistical tests to determine if they are from the same population as the existing database.

The first step is to [ ]+ +(a,b,c)

The final test is to [ ]+ +(a,b,c)

An example of the application of this procedure in WCAP-9401-P-A and is summarized below. In this example, two OFA tests series were run and compared to the WRB-1 correlation which was derived from a series of earlier tests.

The data from the new tests were evaluated with the WRB-1 correlation. The $\overline{M/P}$ values of the new data were evaluated using the procedures described in Ref. (2) and found to be normally distributed.

A comparison of the means was made using the procedures described in Ref. (3) to determine the 95 percent tolerance limits on the sample means of the three data sets. The values used and the resulting limits are given in Table 1.

[

]+ +(a,b,c)

The range and number of test parameters are typically chosen [

]+ The data are statistically evaluated using the methods described above. +(a,b,c)

The new correlations 95/95 limit with the pooled database is determined from the expression.

$$95/95(DNBR)_{min} = \frac{1}{(M/P)_{avg} - K_{95/95}S}$$

where

$(M/P)_{avg}$	=	average $\overline{M/P}$ of the pooled database
S	=	sample standard deviation of the pooled database
$K_{95/95}$	=	tolerance factor from Ref. (5) based on the population of the pooled database

TABLE 1

ANALYSIS OF MEANS FOR WCAP-9401 DATA

$[\quad]^{+(b,c)}$

As indicated, the observed mean $\overline{M/P}$ values fall within the calculated tolerance limits, thus verifying that the OFA data are compatible with the standard R-grid data.

A test of hypothesis that the sample variances for the OFA data and standard R-grid data are sample values from the same population was carried out using the F distribution, Ref. (4). These results are given in the following table:

$[\quad]^{+(b,c)}$

REFERENCES

1. U. S. Atomic Energy Commission, Regulatory Guide 5.22, "Assessment of the Assumption of Normality (Empoying Individual Observed Values)," April, 1974.

2. ANSI N15.15-1974, "American National Standard Assessment of the Assumption of Normality (Employing Individual Observed Values), October, 1973.

3. L. S. Nelson, "Factors for the Analysis of Means," Journal of Quality Technology, Vol. 6, No. 4, Oct., 1974, P. 175

4. P. G. Hoel, "Introduction to Mathematical Statistics," 4th Ed., John Wiley and Sons, Inc., New York, 1971, p. 269.

5. D. B. Owen, "Factors for One-Sided Tolerance Limits and for Variables Sampling Plans," SCR-607, Sandra Corp., March 1963.

SECTION 7.0 FUEL PERFORMANCE AND MATERIAL PROPERTIES MODELS

The Westinghouse PAD3.4 fuel performance code, Reference 1, is the basic tool used by Westinghouse to perform fuel rod mechanical design analyses and to generate fuel temperature and rod internal pressure input to safety analyses. The PAD3.4 performance models have been developed on a best estimate basis, with appropriate model uncertainties derived to assure that adequate conservatism is incorporated in all fuel rod design and safety related calculations. Fuel and cladding material properties models, References 2 and 3 (for ZIRLO^{TM}), used in Westinghouse design and safety evaluations have also been reviewed by NRC.

The Fuel Criteria Evaluation Process will be applied to adjustments of existing PAD fuel performance and material properties models, based on the analysis of new data. This ability to update PAD models as additional performance data becomes available is important to assure that all fuel performance analysis and design calculations appropriately reflect expected behavior. The revised fuel performance models will be established to produce consistent overall agreement between the model prediction and the updated fuel performance data base. The use of performance and material property model uncertainties, typically defined to bound at least 95% of the model data base, assures that analyses performed with the updated PAD3.4 code will remain appropriately conservative.

Modified fuel performance and material property models will employ the [

]$^{+}$ will not be implemented using the Fuel Criteria Evaluation process. +(a,c)
These model revisions will be submitted for explicit NRC review and approval.

This application of the Fuel Criteria Evaluation Process to fuel performance and materials properties models will permit model revisions, including extensions of the range of applicability of the models, without formal NRC review provided that the following conditions are satisfied:

1) The change in the integrated fuel performance code response for

   a) maximum fuel average temperature,
   b) end of life rod internal pressure, and
   c) cladding strain,

   relative to that predicted using the NRC approved PAD3.4 code version for established benchmark cases will not exceed the limits defined in Table 7.1.

2) The change in the predictive capability for each model which is modified, relative to the reference performance model data base for the NRC approved PAD code version, will be within the value specified for each model in Table 7.2. Only those models specified in Table 7.2 will be subject to revision under the Fuel Criteria Evaluation Process.

3) New fuel performance or material property data sets to be used in developing a modified performance or material property model will be evaluated relative to the current NRC approved model prior to incorporation into an updated model development data base. Only those data sets which meet the limits specified in Table 7.2 relative to the current approved model will be used in model adjustments under the Fuel Criteria Evaluation Process.

4) New fuel performance models will not be introduced using the Fuel Criteria Evaluation Process. Any new performance phenomena to be modeled in a revised PAD version will be submitted to the NRC for explicit review and approval. Subsequent to NRC approval of a new model, future revisions to that model will be subject to the Fuel Criteria Evaluation Process.

5) The Fuel Criteria Evaluation Process will not be used to extend the range of applicability of any fuel performance or material property model to burnups [ ]+ beyond the target lead rod average licensed burnup limits approved for Westinghouse fuel by the NRC. +(a,c)

6) The Fuel Criteria Evaluation Process will not be used to extend the applicability of fuel performance or material property models to new materials, where new materials are defined as materials which have not explicitly been reviewed and approved for use in reactors by the NRC.

Documentation of the basis for all model adjustments will be maintained, including appropriate statistical tests for goodness of fit and variability. Uncertainties in the models will be addressed and bounding models (95% bounding) will be established using all available applicable data as required to appropriately incorporate the impact of uncertainties in fuel licensing analyses. Westinghouse will transmit to the NRC for information any fuel performance and material property model modifications implemented under the Fuel Criteria Evaluation Process within six months of implementation.

Benchmark cases have been defined for evaluating the change in the integrated fuel performance code response per the limits identified in Table 7.1. These benchmark cases include [

]+ +(a,c)

[

]+ +(a,c)

Additional fission gas release data for increased Gd_2O_3 concentrations will be reviewed as they become available. If these data confirm that existing fission gas release models are applicable to higher gadolinia concentration fuel, then, under the Fuel Criteria Evaluation Process, these conclusions will be appropriately documented and the applicability of the PAD models will be extended to the higher gadolinia concentrations.

Should the fission gas release data for higher gadolinia concentrations indicate that a model adjustment is required, if the adjusted model satisfies the conditions specific in Tables 7.1 and 7.2, the adjusted model will be documented and implemented as part of the Fuel Criteria Evaluation Process. Should the revised models not satisfy the conditions in Tables 7.1 and 7.2, then an explicit NRC review of the proposed model change would be required prior to implementation.

All fuel rod designs which employ gadolinia concentrations [ ]+ will be shown to satisfy all of the criteria established in Section 4 of the Fuel Criteria Evaluation Process. +(a,c)

References:

1. Weiner, R. A., et al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851-P-A (Proprietary) and WCAP-11873-A (Non-Proprietary), August 1988.

2. Kuchirka, P. J., "Properties of Fuel and Core Component Material," WCAP-9179, Revision 1, July 1978.

3. Davidson, S. L. and Nuhfer, D. L., Ed., "VANTAGE+ Fuel Assembly Reference Core Report," WCAP-12610, June 1990.

4. NRC Memorandum, C. E. Rossi (NRC) to E. P. Rahe, Jr. (W), "Acceptance for Referencing of Licensing Topical Report WCAP-8720 Addenda 3 'Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations/Application for BWR Fuel Analysis,'" September 15, 1986.

5. Ewing, J. H. and Smith, W. L., "Improved Analytical Models used in Westinghouse Fuel Rod Design Computations/Application for BWR Fuel Analysis," WCAP-8720, Addendum 3, January 1983.

6. NS-EPR-2835, "Response to Request for Additional Information on WCAP-8720, Addendum 3," November 4, 1983.

7. NS-NRC-86-3145, "Response to NRC Gadolinia Questions on Westinghouse Topical Report, WCAP-8720, Addenda 3," June 30, 1986.

Table 7.1
Limits for Allowable Change in the Integrated
Fuel Performance Model Response to Model Updates

+(a,c)

Table 7.2
Limits for Allowable Change in the Individual Performance and Material Property Models
Relative to the Reference Model Data Base

+(a,b,c)

SECTION J

Westinghouse Electric Corporation
Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

March 29, 1993
ET-NRC-93-3842

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Attention: R. C. Jones, Reactor Systems Branch Chief, Division of Engineering and System Technology

Subject: Final Responses to Additional Information on WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process Revision 1" [Proprietary]

Reference: (1) Letter from N. J. Liparulo (Westinghouse) to R. C. Jones (NRC), ET-NRC-93-3819 "Final Responses to Additional Information on WCAP-12488, Westinghouse Fuel Criteria Evaluation Process," February 8, 1993. [Proprietary]

Dear Mr. Jones:

After discussions with your staff, the Proprietary information contained in our Reference (1) submittal was reviewed and several sections of the text were reclassified from Westinghouse Proprietary to Non-Proprietary. Enclosed are fifteen (15) copies of the reclassified text in response to Question 3(b) (DNB Test Data) and for the Section 7.0 Supplement (Fuel Performance and Material Properties Models) transmitted previously in Reference (1). This revised submittal replaces the Reference (1) submittal.

This submittal contains Westinghouse proprietary information of trade secrets, commercial or financial information which we consider privileged or confidential pursuant to 10CFR9.5(4). Therefore, it is requested that the Westinghouse proprietary information attached hereto be handled on a confidential basis and be withheld from public disclosure.

This material is for your internal use only and may be used solely for the purpose for which it is submitted. It should not be otherwise used, disclosed, duplicated, or disseminated, in whole or in part, to any other person or organization outside the Office of Nuclear Reactor Regulation without the express prior written approval of Westinghouse.

Correspondence with respect to any Application for Withholding should reference AW-93-433 and should be addressed to N. J. Liparulo, Manager of Regulatory and Legislative Affairs, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

Nicholas J. Liparulo

Nicholas J. Liparulo, Manager
Nuclear Safety and Regulatory Activities

cc: L. E. Phillips - NRC (MS 8E23)
S. L. Wu - NRC (MS 8E23)

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

March 29, 1993
AW-93-433

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Attention: Mr. R. C. Jones, Reactor Systems Branch Chief, Division of Engineering and System Technology

Reference: Letter from N. J. Liparulo (Westinghouse) to R. C. Jones (NRC), ET-NRC-93-3842, dated March 29, 1993

Subject: Final Responses for Additional Information on WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process Revision 1" [Proprietary]

Dear Mr. Jones:

The above referenced letter contains information proprietary to the Westinghouse Electric Corporation.

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

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

Correspondence with respect to the proprietary aspects of this submittal should reference AW-93-433 and should be addressed to the undersigned.

Very truly yours,

Nicholas J. Liparulo, Manager
Nuclear Safety and Regulatory Activities

cc: K. Bohrer/NRC (12H5)

Revision to Response to Question 3(b) on WCAP-12488

Question 3(b) *If new thermal hydraulic tests are performed, what statistical tests will be used to determine if the new DNB data for the new design are from the same population as the existing database? Please provide an example application of the statistical methods that will be used to determine if the new database is from the same population as the existing database. How will the range of test parameters be chosen for the experimental data from the new design and how will it be determined that the number of test data are satisfactory? Also, please provide an example of the statistical methods that will be used to determine a new correlations' 95/95 limit with the "pooled" database.*

Response

The statistical tests performed by Westinghouse [ ]+ in WCAP-9401 using standard statistical tests to [ ]+ The procedural steps are: +(a,b,c)

+(a,b,c)

[ ]+ +(a,b,c)

The ratio of measured to predicted critical heat fluxes, $\overline{M/P}$, are subjected to the three statistical tests to determine if they are from the same population as the existing database.

The first step is to [ ]+ +(a,b,c)

The final test is to [ ]+ +(a,b,c)

An example of the application of this procedure in WCAP-9401-P-A and is summarized below. In this example, two OFA tests series were run and compared to the WRB-1 correlation which was derived from a series of earlier tests.

The data from the new tests were evaluated with the WRB-1 correlation. The $\overline{M/P}$ values of the new data were evaluated using the procedures described in Ref. (2) and found to be normally distributed.

A comparison of the means was made using the procedures described in Ref. (3) to determine the 95 percent tolerance limits on the sample means of the three data sets. The values used and the resulting limits are given in Table 1.

[

.]$^{+}$ $^{+}$(a,b,c)

The range and number of test parameters are typically chosen [

]$^{+}$ The data are statistically evaluated using the methods described above. $^{+}$(a,b,c)

The new correlations 95/95 limit with the pooled database is determined from the expression.

$$95/95(DNBR)_{min} = \frac{1}{(M/P)_{avg} - K_{95/95}S}$$

where

$(M/P)_{avg}$	=	average $\overline{M/P}$ of the pooled database
S	=	sample standard deviation of the pooled database
$K_{95/95}$	=	tolerance factor from Ref. (5) based on the population of the pooled database

TABLE 1

ANALYSIS OF MEANS FOR WCAP-9401 DATA

[ ]+(b,c)

As indicated, the observed mean $\overline{M/P}$ values fall within the calculated tolerance limits, thus verifying that the OFA data are compatible with the standard R-grid data.

A test of hypothesis that the sample variances for the OFA data and standard R-grid data are sample values from the same population was carried out using the F distribution, Ref. (4). These results are given in the following table:

[ ]+(b,c)

REFERENCES

1. U. S. Atomic Energy Commission, Regulatory Guide 5.22, "Assessment of the Assumption of Normality (Empoying Individual Observed Values)," April, 1974.

2. ANSI N15.15-1974, "American National Standard Assessment of the Assumption of Normality (Employing Individual Observed Values), October, 1973.

3. L. S. Nelson, "Factors for the Analysis of Means," Journal of Quality Technology, Vol. 6, No. 4, Oct., 1974, P. 175

4. P. G. Hoel, "Introduction to Mathematical Statistics," 4th Ed., John Wiley and Sons, Inc., New York, 1971, p. 269.

5. D. B. Owen, "Factors for One-Sided Tolerance Limits and for Variables Sampling Plans," SCR-607, Sandra Corp., March 1963.

SECTION 7.0 FUEL PERFORMANCE AND MATERIAL PROPERTIES MODELS

The Westinghouse PAD3.4 fuel performance code, Reference 1, is the basic tool used by Westinghouse to perform fuel rod mechanical design analyses and to generate fuel temperature and rod internal pressure input to safety analyses. The PAD3.4 performance models have been developed on a best estimate basis, with appropriate model uncertainties derived to assure that adequate conservatism is incorporated in all fuel rod design and safety related calculations. Fuel and cladding material properties models, References 2 and 3 (for ZIRLO™), used in Westinghouse design and safety evaluations have also been reviewed by NRC.

The Fuel Criteria Evaluation Process will be applied to adjustments of existing PAD fuel performance and material properties models, based on the analysis of new data. This ability to update PAD models as additional performance data becomes available is important to assure that all fuel performance analysis and design calculations appropriately reflect expected behavior. The revised fuel performance models will be established to produce consistent overall agreement between the model prediction and the updated fuel performance data base. The use of performance and material property model uncertainties, typically defined to bound at least 95% of the model data base, assures that analyses performed with the updated PAD3.4 code will remain appropriately conservative.

Modified fuel performance and material property models will employ the [

]$^{+}$ will not be implemented using the Fuel Criteria Evaluation process. +(a,c)
These model revisions will be submitted for explicit NRC review and approval.

This application of the Fuel Criteria Evaluation Process to fuel performance and materials properties models will permit model revisions, including extensions of the range of applicability of the models, without formal NRC review provided that the following conditions are satisfied:

1) The change in the integrated fuel performance code response for

   a) maximum fuel average temperature,
   b) end of life rod internal pressure, and
   c) cladding strain,

   relative to that predicted using the NRC approved PAD3.4 code version for established benchmark cases will not exceed the limits defined in Table 7.1.

2) The change in the predictive capability for each model which is modified, relative to the reference performance model data base for the NRC approved PAD code version, will be within the value specified for each model in Table 7.2. Only those models specified in Table 7.2 will be subject to revision under the Fuel Criteria Evaluation Process.

3) New fuel performance or material property data sets to be used in developing a modified performance or material property model will be evaluated relative to the current NRC approved model prior to incorporation into an updated model development data base. Only those data sets which meet the limits specified in Table 7.2 relative to the current approved model will be used in model adjustments under the Fuel Criteria Evaluation Process.

4) New fuel performance models will not be introduced using the Fuel Criteria Evaluation Process. Any new performance phenomena to be modeled in a revised PAD version will be submitted to the NRC for explicit review and approval. Subsequent to NRC approval of a new model, future revisions to that model will be subject to the Fuel Criteria Evaluation Process.

5) The Fuel Criteria Evaluation Process will not be used to extend the range of applicability of any fuel performance or material property model to burnups [ ]$^+$ beyond the target lead rod average licensed burnup limits approved for Westinghouse fuel by the NRC. $^{+}$(a,c)

6) The Fuel Criteria Evaluation Process will not be used to extend the applicability of fuel performance or material property models to new materials, where new materials are defined as materials which have not explicitly been reviewed and approved for use in reactors by the NRC.

Documentation of the basis for all model adjustments will be maintained, including appropriate statistical tests for goodness of fit and variability. Uncertainties in the models will be addressed and bounding models (95% bounding) will be established using all available applicable data as required to appropriately incorporate the impact of uncertainties in fuel licensing analyses. Westinghouse will transmit to the NRC for information any fuel performance and material property model modifications implemented under the Fuel Criteria Evaluation Process within six months of implementation.

Benchmark cases have been defined for evaluating the change in the integrated fuel performance code response per the limits identified in Table 7.1. These benchmark cases include [

]$^+$ $^{+}$(a,c)

[

]$^+$ $^+$(a,c)

Additional fission gas release data for increased Gd_2O_3 concentrations will be reviewed as they become available. If these data confirm that existing fission gas release models are applicable to higher gadolinia concentration fuel, then, under the Fuel Criteria Evaluation Process, these conclusions will be appropriately documented and the applicability of the PAD models will be extended to the higher gadolinia concentrations.

Should the fission gas release data for higher gadolinia concentrations indicate that a model adjustment is required, if the adjusted model satisfies the conditions specific in Tables 7.1 and 7.2, the adjusted model will be documented and implemented as part of the Fuel Criteria Evaluation Process. Should the revised models not satisfy the conditions in Tables 7.1 and 7.2, then an explicit NRC review of the proposed model change would be required prior to implementation.

All fuel rod designs which employ gadolinia concentrations [ ]$^+$ will be $^+$(a,c) shown to satisfy all of the criteria established in Section 4 of the Fuel Criteria Evaluation Process.

References:

1. Weiner, R. A., et al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851-P-A (Proprietary) and WCAP-11873-A (Non-Proprietary), August 1988.

2. Kuchirka, P. J., "Properties of Fuel and Core Component Material," WCAP-9179, Revision 1, July 1978.

3. Davidson, S. L. and Nuhfer, D. L., Ed., "VANTAGE+ Fuel Assembly Reference Core Report," WCAP-12610, June 1990.

4. NRC Memorandum, C. E. Rossi (NRC) to E. P. Rahe, Jr. (W), "Acceptance for Referencing of Licensing Topical Report WCAP-8720 Addenda 3 'Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations/Application for BWR Fuel Analysis,'" September 15, 1986.

5. Ewing, J. H. and Smith, W. L., "Improved Analytical Models used in Westinghouse Fuel Rod Design Computations/Application for BWR Fuel Analysis," WCAP-8720, Addendum 3, January 1983.

6. NS-EPR-2835, "Response to Request for Additional Information on WCAP-8720, Addendum 3," November 4, 1983.

7. NS-NRC-86-3145, "Response to NRC Gadolinia Questions on Westinghouse Topical Report, WCAP-8720, Addenda 3," June 30, 1986.

Table 7.1
Limits for Allowable Change in the Integrated
Fuel Performance Model Response to Model Updates

$^{+}$(a,c)

Table 7.2
Limits for Allowable Change in the Individual Performance and Material Property Models
Relative to the Reference Model Data Base

+(a,b,c)

SECTION K

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

August 25, 1993
ET-NRC-93-3593

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Attention: R. C. Jones, Reactor Systems Branch Chief, Division of Engineering and System Technology

Subject: "Westinghouse Fuel Criteria Evaluation Process" - WCAP-12488, Fuel Clad Fretting Wear Criteria

Dear Mr. Jones:

As a result of discussions with your Staff and PNL on August 20th, enclosed are fifteen (15) copies of the revised Fuel Clad Fretting Wear Criteria contained in WCAP-12488. The Design Evaluation was amended to include as an alternate, flow induced vibration testing of a single fuel assembly. This revision should provide the information needed from Westinghouse to complete your review of WCAP-12488.

This submittal contains Westinghouse proprietary information of trade secrets, commercial or financial information which we consider privileged or confidential pursuant to 10CFR9.5(4). Therefore, it is requested that the Westinghouse proprietary information attached hereto be handled on a confidential basis and be withheld from public disclosure.

This material is for your internal use only and may be used solely for the purpose for which it is submitted. It should not be otherwise used, disclosed, duplicated, or disseminated, in whole or in part, to any other person or organization outside the Office of Nuclear Reactor Regulation without the express prior written approval of Westinghouse.

Correspondence with respect to any Application for Withholding should reference AW-93-513 and should be addressed to N. J. Liparulo, Manager of Regulatory and Legislative Affairs, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

Nicholas J. Liparulo

Nicholas J. Liparulo, Manager
Nuclear Safety and Regulatory Activities

cc: L. Barnett - NRR (MIPA)
L. E. Phillips - NRR (SRXB)

Westinghouse Electric Corporation
Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

August 25, 1993
AW-93-513

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Attention: Mr. R. C. Jones, Reactor Systems Branch Chief, Division of Engineering and System Technology

Reference: Letter from N. J. Liparulo (Westinghouse) to R. C. Jones (NRC), ET-NRC-93-3593, dated August 25, 1993

Subject: "Westinghouse Fuel Criteria Evaluation Process" - WCAP-12488, Fuel Clad Fretting Wear Criteria

Dear Mr. Jones:

The above referenced letter contains information proprietary to the Westinghouse Electric Corporation.

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

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

Correspondence with respect to the proprietary aspects of this submittal should reference AW-93-513 and should be addressed to the undersigned.

Very truly yours,

Nicholas J. Liparulo

Nicholas J. Liparulo, Manager
Nuclear Safety and Regulatory Activities

cc: K. Bohrer/NRC 12H5
Office of the General Counsel, NRC

Design Category - Fuel System Damage and Fuel Rod Failure Criteria

Design Parameter - Fuel Clad Fretting Wear

Design Basis - The fuel system will not be damaged due to fuel clad fretting.

Acceptance Limit - Westinghouse uses a wall thickness reduction of [ ]$^+$ percent, Reference 1, as a general guide in evaluating clad imperfections, including fretting wear marks. Furthermore, both clad stress and fatigue acceptance limits must be met. $^+$(a,c)

Design Evaluation - The fretting wear evaluation is performed: (1) on the basis of experimental data [ ]$^+$ or (2) using an [$^+$(a,b,c)
]$^+$ flow tests, References $^+$(a,b,c)
2 and 3, or (3) vibration [
]$^+$ test to assure that the assembly does not [$^+$(a,b,c)
]$^+$ and/or (4) using the calculated [$^+$(a,c)
]$^+$ $^+$(a,c)

Reference(s)

1. Davidson, S. L. (ed.) et al., "Extended Burnup Evaluation of Westinghouse Fuel, "WCAP-10125-P-A, December 1985.

2. Davidson, S. L., and Iorii, J. A., "Reference Core Report 17x17 Optimized Fuel Assembly," WCAP-9500-A, May 1982.

3. Davidson, S. L., Kramer, W. R. (Eds.) "Reference Core Report VANTAGE 5 Fuel Assembly," WCA-10444-P-A, September 1985.

SECTION L

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

February 2, 1994

Mr. E. H. Novendstern, Manager
Nuclear Manufacturing Division
Westinghouse Electric Corporation
P. O. Box 355
Pittsburgh, PA 15230

Dear Mr. Novendstern:

SUBJECT: SECOND REQUEST FOR ADDITIONAL INFORMATION ON WCAP-12488, "WESTINGHOUSE FUEL CRITERIA EVALUATION PROCESS," (TAC NO. M77257)

We have nearly completed the review of the Westinghouse Topical Report WCAP-12488 "Westinghouse Fuel Criteria Evaluation Process". However, a concern was raised due to the use of U02-Gd203 fuel as described in the enclosure. You are requested to respond to the question as expeditiously as possible in order for us to complete the review. Should you have any question regarding this request for information, please contact Mr. S. L. Wu of my staff at (301)504-3284.

Sincerely,

Robert C. Jones, Chief
Reactor Systems Branch
Division of Systems Safety and Analysis

Enclosure:
Request for Information

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

ENCLOSURE

In Section 7.0 "Fuel Performance and Material Properties Models" described in the Final Responses to Additional Information on WCAP-12488 dated March 29, 1993, Westinghouse described a mechanical evaluation process to extend the U02-Gd203 fuel to 10 w/o Gd203. Since the use of U02-Gd203 also involves neutronic calculations, please provide a description of the neutronic evaluation process for U02-Gd203 fuel. In addition, Westinghouse should also address the implications and requirements if a licensee elects to use its own neutronic methodology to analyze Westinghouse U02-Gd203 fuel.

SECTION M

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

February 28, 1994
NTD-NRC-94-4073

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Attention: R. C. Jones, Chief
Reactor Systems Branch
Division of System Technology

Subject: Response to Second Request for Additional Information on WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process" [Proprietary]

Reference (1): Letter from R. C. Jones (NRC) to E. H. Novendstern (Westinghouse), "Second Request for Additional Information on WCAP-12488, 'Westinghouse Fuel Criteria Evaluation Process,'" (TAC No. M77257), dated February 2, 1994.

Dear Mr. Jones:

Enclosed are fifteen (15) copies [Proprietary] in response for additional information requested in Reference 1. This response should close out your review of WCAP-12488.

This submittal contains Westinghouse proprietary information of trade secrets, commercial or financial information which we consider privileged or confidential pursuant to 10 CFR 9.5(4). Therefore, it is requested that the Westinghouse proprietary information attached hereto be handled on a confidential basis and be withheld from public disclosure.

This material is for your internal use only and may be used solely for the purpose for which it is submitted. It should not be otherwise used, disclosed, duplicated, or disseminated, in whole or in part, to any other person or organization outside the Office of Nuclear Reactor Regulation without the expressed prior written approval of Westinghouse.

Correspondence with respect to any Application for Withholding should reference AW-94-598 and should be addressed to N. J. Liparulo, Manager of Nuclear Safety and Regulatory Activities, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

Nicholas J. Liparulo, Manager
Nuclear Safety and Regulatory Activities

cc: L. W. Barnett, NRC (MS 12H5)
L. E. Phillips, NRC (MS8E23)
S. L. Wu, NRC (MS 8E23)

Westinghouse Electric Corporation
Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

February 28, 1994
AW-94-598

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Attention: Mr. R. C. Jones, Reactor Systems Branch Chief, Division of Engineering and System Technology

Reference: Letter from N. J. Liparulo (Westinghouse) to R. C. Jones (NRC), NTD-NRC-94-4073, February 28, 1994

Subject: Response to Second Request for Additional Information on WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process," [Proprietary]

Dear Mr. Jones:

The above referenced letter contains information proprietary to the Westinghouse Electric Corporation.

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

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

Correspondence with respect to the proprietary aspects of this submittal should reference AW-94-598 and should be addressed to the undersigned.

Very truly yours,

Nicholas J. Liparulo, Manager
Nuclear Safety and Regulatory Activities

cc: Kevin Bohrer / NRC (12H5)

NRC Request For Additional Information

In Section 7.0 "Fuel Performance and Material Properties Models" described in the Final Responses to Additional Information on WCAP-12488 dated March 29, 1993, Westinghouse described a mechanical evaluation process to extend the UO_2-Gd_2O_3 fuel to 10 w/o Gd_2O_3. Since the use of UO_2-Gd_2O_3 also involves neutronic calculations, please provide a description of the neutronic evaluation process for UO_2-Gd_2O_3 fuel. In addition, Westinghouse should also address the implications and requirements if a licensee elects to use its own neutronic methodology to analyze Westinghouse UO_2-Gd_2O_3 fuel.

Response

The NRC approved Westinghouse neutronic methods, described in Reference 1, have a very strong theoretical basis since the methodology does not employ empirical adjustments in code predictions. Predicted results are not effected by the usage of an individual fuel feature such as Gadolinia concentrations > 6 w/o. The range of applicability of the methodology is utilized throughout LWR applications and Westinghouse expected the performance of the code system to be accurate for higher Gadolinia concentrations.

The use of Gadolinia concentration > 6 w/o is an example of how the Westinghouse Fuel Criteria Evaluation Process (FCEP) can be used to include performance verification and demonstrate the applicability of current NRC approved methods/codes. Uncertainties in the analysis would be considered in this performance verification of the methodology. The following is a description of the evaluation process to be used for the application of Gadolinia fuel.

As a precedent for this neutronic evaluation process, Westinghouse has previously used NRC approved methods/models for Gadolinia fuel, Reference 1. To extend the applicability of these NRC approved methods/models to higher concentrations of Gadolinia, predicted model performance will be verified using available [

]$^+$ $^+$(a,c)

These measurements typically would include:

- [
-
-]$^+$ $^+$(a,b,c)

The above performance verification would be examined relative to previously approved results given in Reference 3. Evaluations would confirm that the methodology's performance is well within the current performance data base of hundreds of previously designed/operated cycles of plants using a wide variety of Westinghouse fuel products. This statistical evaluation assumes that the various uncertainty components are independent and the variance is summed to determine an overall composite uncertainty. [

]$^+$ $^+$(a,c)

The applicable nuclear design criteria given in the FCEP, WCAP-12488, Pages 35 - 42 will be examined to insure they remain satisfied with the increased Gadolinia concentration.

If all items in the above process are satisfied and the previously approved methods, models, and uncertainties apply, then the FCEP can be used to address the increase in Gadolinia concentration.
If not, an NRC submittal is required.

Licensee Performing Gadolinia Neutronic Analysis

A number of licensees have signed agreements to use Westinghouse design methods/codes. Upon completion of the performance verification described in the paragraphs above, these licensees can utilize the NRC approved methods/codes for neutronics predictions with Gadolinia concentration greater than 6 w/o.

When the licensee is not using Westinghouse design methods/codes, the licensing applicability of codes and methodology used to model Gadolinia fuel is the responsibility of the licensee. Upon NRC approval of the licensees non-Westinghouse neutronic methods/codes, results can be used in conjunction with the FCEP process to assure all criteria are satisfied.

References

1. Davidson, S. L., (ed.) et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272-P-A, July, 1985.

2. Harris, A. J. et al., "A Description of the Nuclear Design and Analysis Programs for Boiling Water Reactors," WCAP-10106-P-A, June 1982.

3. Nguyen, T. Q., et al., "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," WCAP-11596-P-A, June 1988.

SECTION N

Westinghouse Electric Corporation
Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

March 17, 1994
NTD-NRC-94-4080

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Attention: R. C. Jones, Chief
Reactor Systems Branch
Division of System Technology

Subject: Revised Response to Second Request for Additional Information on WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process" [Proprietary]

Reference (1): Letter from R. C. Jones (NRC) to E. H. Novendstern (Westinghouse), "Second Request for Additional Information on WCAP-12488, 'Westinghouse Fuel Criteria Evaluation Process,'" (TAC No. M77257), dated February 2, 1994

(2): Letter from N. J. Liparulo (Westinghouse) to R. C. Jones (NRC), NTD-NRC-94-4073, February 28, 1994

Dear Mr. Jones:

Enclosed are fifteen (15) copies [Proprietary] in response for additional information requested in Reference 1. This response supersedes our original response, Reference 2, as a result of discussions with your staff, and should close out your review of WCAP-12488.

This submittal contains Westinghouse proprietary information of trade secrets, commercial or financial information which we consider privileged or confidential pursuant to 10 CFR 9.5(4). Therefore, it is requested that the Westinghouse proprietary information attached hereto be handled on a confidential basis and be withheld from public disclosure.

This material is for your internal use only and may be used solely for the purpose for which it is submitted. It should not be otherwise used, disclosed, duplicated, or disseminated, in whole or in part, to any other person or organization outside the Office of Nuclear Reactor Regulation without the expressed prior written approval of Westinghouse.

Correspondence with respect to any Application for Withholding should reference AW-94-601 and should be addressed to N. J. Liparulo, Manager of Nuclear Safety and Regulatory Activities, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

Nicholas J. Liparulo, Manager
Nuclear Safety and Regulatory Activities

cc: L. W. Barnett, NRC (MS 12H5)
L. E. Phillips, NRC (MS8E23)
S. L. Wu, NRC (MS 8E23)

/ssh

Westinghouse
Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

March 17, 1994
AW-94-601

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Attention: Mr. R. C. Jones, Reactor Systems Branch Chief, Division of Engineering and System Technology

Reference: Letter from N. J. Liparulo (Westinghouse) to R. C. Jones (NRC), NTD-NRC-94-4080, March 17, 1994

Subject: Revised Response to Second Request for Additional Information on WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process," [Proprietary]

Dear Mr. Jones:

The above referenced letter contains information proprietary to the Westinghouse Electric Corporation.

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

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

Correspondence with respect to the proprietary aspects of this submittal should reference AW-94-601 and should be addressed to the undersigned.

Very truly yours,

Nicholas J. Liparulo, Manager
Nuclear Safety and Regulatory Activities

cc: Kevin Bohrer / NRC (12H5)

NRC Request For Additional Information
In Section 7.0 "Fuel Performance and Material Properties Models" described in the Final Responses to Additional Information on WCAP-12488 dated March 29, 1993, Westinghouse described a mechanical evaluation process to extend the UO_2-Gd_2O_3 fuel to 10 w/o Gd_2O_3. Since the use of UO_2-Gd_2O_3 also involves neutronic calculations, please provide a description of the neutronic evaluation process for UO_2-Gd_2O_3 fuel. In addition, Westinghouse should also address the implications and requirements if a licensee elects to use its own neutronic methodology to analyze Westinghouse UO_2-Gd_2O_3 fuel.

Response

The NRC approved Westinghouse neutronic methods, described in Reference 1, have a very strong theoretical basis since the methodology does not employ empirical adjustments in code predictions. Predicted results are not effected by the usage of an individual fuel feature such as Gadolinia concentrations > 6 w/o. The range of applicability of the methodology is utilized throughout LWR applications and Westinghouse expected the performance of the code system to be accurate for higher Gadolinia concentrations.

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As a precedent for this neutronic evaluation process, Westinghouse has previously used NRC approved methods/models for Gadolinia fuel, Reference 1. To extend the applicability of these NRC approved methods/models to higher concentrations of Gadolinia, predicted model performance will be verified using available [

]$^+$ $^+$(a,c)

These measurements typically would include:
- [
-
-]$^+$ $^+$(a,b,c)

The above performance verification would be examined relative to previously approved results given in Reference 3. Evaluations would confirm that the methodology's performance is well within the current performance data base of hundreds of previously designed/operated cycles of plants using a wide variety of Westinghouse fuel products. This statistical evaluation assumes that the various uncertainty components are independent and the variance is summed to determine an overall composite uncertainty. [

]$^+$ $^+$(a,c)

The applicable nuclear design criteria given in the FCEP, WCAP-12488, Pages 35 - 42 will be examined to insure they remain satisfied with the increased Gadolinia concentration.

If all items in the above process are satisfied and the previously approved methods, models, and uncertainties apply, then the FCEP can be used to address the increase in Gadolinia concentration.
If not, an NRC submittal is required.

Licensee Performing Gadolinia Neutronic Analysis

A number of licensees have signed agreements to use Westinghouse design methods/codes. Upon completion of the performance verification described in the paragraphs above, these licensees can utilize the NRC approved methods/codes for neutronics predictions with Gadolinia concentration greater than 6 w/o.

When the licensee is not using Westinghouse design methods/codes, the licensing applicability of codes and methodology used to model Gadolinia fuel is the responsibility of the licensee. Upon NRC approval of the licensees non-Westinghouse neutronic methods/codes, results can be used in conjunction with the FCEP process to assure all criteria are satisfied.

Modification to WCAP-12488

After review of the above response, Westinghouse believes it is appropriate to include the concept described in the above two paragraphs in the main body of the report. Therefore, Westinghouse will add the following paragraph after the first paragraph in Section 3.2, Application, page 4.

As stated in the Section 1 of this report, the fuel criteria are evaluated using NRC approved codes and methods. These codes and methods are referenced throughout the report, and the evaluations that are described are usually done solely by Westinghouse. However, some licensees, who have signed agreements to use Westinghouse design methods/codes, desire to do portions of the evaluations themselves. Upon the licensee obtaining the necessary NRC approval, they can utilize the methods/codes for the areas in which they have received NRC approval. Additionally, when the licensee is not using Westinghouse design codes/methods, the licensing applicability of codes and methodology used to model the fuel change is the responsibility of the licensee. Upon NRC approval of the licensee's non-Westinghouse neutronic methods/codes, results can be used on conjunction with the FCEP process to assure all criteria are satisfied.

References

1. Davidson, S. L., (ed.) et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272-P-A, July, 1985.

2. Harris, A. J. et al., "A Description of the Nuclear Design and Analysis Programs for Boiling Water Reactors," WCAP-10106-P-A, June 1982.

3. Nguyen, T. Q., et al., "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," WCAP-11596-P-A, June 1988.

SECTION O

Westinghouse
Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

May 20, 1994
NTD-NRC-94-4137

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Attention: T. E. Collins, Acting Chief
Reactor Systems Branch
Division of System Technology

Subject: WRB-1 Correlation Applicability

Dear Mr. Collins:

During a meeting with your Staff concerning our topical report, WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process," on April 21, 1994, in Rockville, the applicability of the WRB-1 DNB correlation to the 17x17 Standard R-grid, Optimized Fuel Assembly (OFA) and VANTAGE 5H grid designs was discussed. Attached is supplemental background information that Westinghouse agreed to supply to the NRC at this meeting. This information clearly demonstrates that the WRB-1 correlation is applicable to the above grid designs. Where appropriate the generic topical and NRC SER are noted to substantiate the use of the WRB-1 DNB correlation for the various Westinghouse grid designs.

This submittal contains Westinghouse proprietary information of trade secrets, commercial or financial information which we consider privileged or confidential pursuant to 10 CFR 9.5(4). Therefore, it is requested that the Westinghouse proprietary information attached hereto be handled on a confidential basis and be withheld from public disclosure.

This material is for your internal use only and may be used solely for the purpose for which it is submitted. It should not be otherwise used, disclosed, duplicated, or disseminated, in whole or in part, to any other person or organization outside the Office of Nuclear Reactor Regulation without the expressed prior written approval of Westinghouse.

Correspondence with respect to any Application for Withholding should reference AW-94-634 and should be addressed to N. J. Liparulo, Manager of Nuclear Safety Regulatory and Licensing Activities, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

Nicholas J. Liparulo, Manager
Nuclear Safety Regulatory and Licensing Activities

cc: L. W. Barnett, NRC (MS 12H5)
L. E. Phillips, NRC (MS8E23)
S. L. Wu, NRC (MS 8E23)

/ssh

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

May 20, 1994
AW-94-634

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Attention: Mr. T. E. Collins, Reactor Systems Branch Acting Chief, Division of Engineering and System Technology

Reference: Letter from N. J. Liparulo (Westinghouse) to T. E. Collins (NRC), NTD-NRC-94-4137, May 20, 1994

Subject: WRB-1 DNB Correlation Applicability

Dear Mr. Collins:

The above referenced letter contains information proprietary to the Westinghouse Electric Corporation.

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

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

Correspondence with respect to the proprietary aspects of this submittal should reference AW-94-634 and should be addressed to the undersigned.

Very truly yours,

H. Sepp /for

Nicholas J. Liparulo, Manager
Nuclear Safety Regulatory and Licensing Activities

cc: Kevin Bohrer / NRC (12H5)

Grid Geometries - see attached Figure

Standard Grid - WCAP-8762 - submitted 1976, NRC SER 1978. For the 17x17 configuration, the R-grid (or standard grid) was utilized in testing; the rod diameter is 0.374 inches. At this time, OFA had not been developed.

OFA Grid - WCAP-9401 - submitted 1979, NRC SER 1981. This report contained the description/results. The pressure drop for this grid is considerably greater than the Standard or V5H grid. WRB-1 was shown to be applicable to OFA (and also still applicable to Standard). SER states (page B-22)

"4.0 Regulatory Position

We have reviewed WCAP-9401 and additional supporting material ... As a result of our review, we conclude the following:

1. The WRB-1 CHF correlation is an acceptable correlation for use in the thermal-hydraulic analysis of OFA.

2. The DNBR limit of 1.17 is an acceptable thermal design limit for the OFA.

..."

VANTAGE 5H Grid - DNB testing of the V5H grid design has been performed as part of the AP600 project. [

]+ +(a,b,c)

(a,c)

SECTION P

Westinghouse
Electric Corporation

Nuclear Manufacturing
Divisions

Box 355
Pittsburgh Pennsylvania 15230-0355

July 1, 1994
NTD-NRC-94-4185

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Attention: L. E. Phillips, Reactor Systems Branch Acting Chief, Division of Engineering and Systems Technology

Subject: Revised Responses to Request for Additional Information on WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process" [Proprietary] Provided in ET-NRC-92-3702

Reference: (1) Letter from R. C. Jones (NRC) to E. H. Novendstern (Westinghouse), Request for Additional Information on WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process," April 24, 1992.

(2) Letter from N. J. Liparulo (Westinghouse) to R. C. Jones (NRC), "Responses to Request for Additional Information on WCAP-12488, Westinghouse Fuel Criteria Evaluation Process," ET-NRC-92-3702, June 8, 1992.

Dear Mr. Phillips:

Enclosed are six (6) copies of the Westinghouse revised responses [Proprietary] to your request previously issued for additional information, Reference 1, on WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process." This revision is highlighted on Page 3 of the responses and considers both the grid and vane orientation in the development of the CHF correlation. This submittal supercedes the information presented in Reference (2). Your timely review of this revision is requested in order to complete the review of the topical.

This submittal contains Westinghouse proprietary information of trade secrets, commercial or financial information which we consider privileged or confidential pursuant to 10CFR9.5(4). Therefore, it is requested that the Westinghouse proprietary information attached hereto be handled on a confidential basis and be withheld from public disclosure.

This material is for your internal use only and may be used solely for the purpose for which it is submitted. It should not be otherwise used, disclosed, duplicated, or disseminated, in whole or in part, to any other person or organization outside the Office of Nuclear Reactor Regulation without the express written approval of Westinghouse.

Correspondence with respect to any Application for Withholding should reference AW-94-655 and should be addressed to N. J. Liparulo, Manager of Nuclear Safety Regulatory and Licensing Activities, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

Nicholas J. Liparulo, Manager
Nuclear Safety Regulatory and Licensing Activities

cc: L. Barnett - NRC (MS 12H5)
S. L. Wu - NRC (MS 8E23)
H. Richings - NRC (MS 8E23)

/ssh

Westinghouse
Electric Corporation

Nuclear Manufacturing
Divisions

Box 355
Pittsburgh Pennsylvania 15230-0355

July 1, 1994
AW-94-655

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Attention: Mr. L. E. Phillips, Reactor Systems Branch Acting Chief, Division of Engineering and Systems Technology

Reference: Letter from N. J. Liparulo (Westinghouse) to L. E. Phillips (NRC), ET-NRC-94-4185, dated July 1, 1994

Subject: Revised Responses to Request for Additional Information on WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process," [Proprietary] Provided in ET-NRC-92-3702

Dear Mr. Phillips:

The above referenced letter contains information proprietary to the Westinghouse Electric Corporation.

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

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

Correspondence with respect to the proprietary aspects of this submittal should reference AW-94-655 and should be addressed to the undersigned.

Very truly yours,

Nicholas J. Liparulo, Manager
Nuclear Safety Regulatory and Licensing Activities

cc: Kevin Bohrer/NRC (12H5)

Westinghouse Responses To NRC Questions On The Westinghouse Fuel Criteria Evaluation Process (FCEP) WCAP-12488

CATEGORY 1

Question 1. *The section entitled "Fuel Assembly Structural Response to Seismic/LOCA Loads" (page 34) states that either Best Estimate or Appendix K models may be used when allowable spacer grid loads are exceeded to demonstrate that requirements for core coolability and control rod insertability for safe shutdown are met. Have the best estimate models been reviewed and approved by the NRC? Please elaborate and be more specific on how the approved best estimate models will be used.*

RESPONSE QUESTION 1

The SECY-83-472 best estimate approach has been approved by the NRC. This approach is not a true best estimate approach but was developed to address concerns for Westinghouse plants which have upper plenum injection. These plants are primarily of the Westinghouse two (2) loop design, which are much older than the current generation of Westinghouse plants and many are not required to include seismic loads in the structural analysis of the fuel. Therefore, grid deformation is not an issue for these plants. Further, Westinghouse does not currently plan to extend the SECY-83-472 best estimate approach to other designs. Therefore, the issue for modeling of deformed grids does not apply.

The new Westinghouse Best Estimate models which are being developed under the provision of 10CFR50.46(a)(1)(i), as amended in October 1988, are in development and have not received NRC approval. The FCEP approach would only account for best estimate effects for plants licensed with NRC approved best estimate LOCA models. Since these models are still under NRC review and application to operating Westinghouse PWRs has not yet begun, the issue of how to model grid deformation has not been addressed. When grid deformation must be addressed for application of best estimate technology, Westinghouse would develop the necessary models at that time and submit the grid deformation/core coolability methodology to the NRC as part of that plant's licensing application.

CATEGORY 2

Question 1. *Additional criteria are needed to define those conditions when a new design will or will not be submitted to NRC for review and approval. Please provide these additional criteria. These additional criteria should also address the introduction of new materials in a design. Of particular concern to NRC are those new designs/materials that significantly impact a) the results of design basis accidents, b) the applicability of NRC approved evaluation models, and/or c) the applicability of operating/design limits [with the exception of DNBR limits as discussed in the subject report], compared to previously approved fuel designs. Please provide examples of design/material changes for which an NRC review would and would not be required based on these proposed criteria.*

RESPONSE QUESTION 1

When a new design or material change is introduced, it is possible that the criteria may not comply with those contained in WCAP-12488. Should this happen, that aspect will be submitted to the NRC for separate review. Subsequent to the NRC approval, the new criteria will be used without future submittals to the NRC.

The application of the Fuel Criteria Evaluation Process is intended for fuel mechanical design changes which are extensions/minor changes to existing fuel assembly/component designs previously reviewed and accepted by the NRC provided that all the established criteria in WCAP-12488 are satisfied. An example of this is the VANTAGE 5H fuel assembly. In this example, the criteria discussed in WCAP-12488 were addressed to insure the acceptability of this design change. It was shown using NRC approved methods/models that compared to the previously approved advanced fuel assembly designs the VANTAGE 5H design change did not impact the design basis incidents or operating/design limits.

The application of the Fuel Criteria Evaluation Process is intended for designs which employ materials and materials properties previously reviewed and accepted by the NRC. Designs which employ materials addressed in WCAP-9179 do not require additional materials specific review by the NRC, provided that all of the established criteria in WCAP-12488 are satisfied. Examples of materials for which specific licensing would not be required under this process include improved Zircaloy-4 cladding and low cobalt bottom nozzles. In both of these examples, the materials which are being used have compositions which lie within the ranges specified for those materials in WCAP-9179. An example of a materials change that would require explicit NRC review is the implementation of a new cladding alloy, such as ZIRLO™ cladding. Although ZIRLO™ is similar to Zircaloy, the criteria of acceptance (10CFR50.46 and 10CFR50 Appendix K) are specifically identified as appropriate for Zircaloy clad fuel. Therefore, an NRC review was required and exemptions were obtained to allow application of those criteria to ZIRLO™ clad fuel. However, once a generic review of ZIRLO™ cladding material is completed by the NRC, new fuel designs which employ ZIRLO™ cladding would not require specific NRC review, provided that all of the established criteria in WCAP-12488 are satisfied.

Question 2. *In Section 3.1 of the submittal it is stated that utilities may supply portions of the calculations, using codes similar to those used by Westinghouse, to support a specific design application. This implies that utility models/codes and calculational methods will be used in some instances. What criteria will be used to determine that the utility codes and analyses are applicable and acceptable to the specific design. Who is responsible for making this determination and how is it determined that the utility has the expertise for making these analyses?*

RESPONSE TO QUESTION 2

The criterion used by Westinghouse to determine if utility codes and analyses are applicable and acceptable to a specific design is that utilities who provide analyses of record must have them explicitly approved by the NRC.

This would be done consistent with established precedent, such as the licensing of Northeast Utilities ("Physics Methodology for PWR Reload Design," NUSCO-152, August 1986) for use of Westinghouse Nuclear Design Technology, or other such arrangements established in the future between Westinghouse and the NRC.

Question 3. *The following questions are related to thermal-hydraulic core analyses:*

(a) *When an existing CHF correlation and DNBR limit, i.e., one that was developed for an existing design, is being proposed for application to a new design, what criteria or tests will be applied to the geometric and correlation parameters to determine that the new assembly is similar to or bracketed by "the test data and correlation parameters"? Conversely, what criteria or tests will be used to determine that thermal-hydraulic and CHF tests are needed for a new design? Please provide an example where a new design is determined to be bracketed by existing test data and correlation parameters and a second example where the new designs not bracketed by existing DNB data and correlation parameters.*

(b) *If new thermal hydraulic tests are performed, what statistical tests will be used to determine if the new DNB data for the new design are from the same population as the existing database? Please provide an example application of the statistical methods that will be used to determine if the new database is from the same population as the existing database. How will the range of test parameters be chosen for the experimental data from the new design and how will it be determined that the number of test data are satisfactory? Also, please provide an example of the statistical methods that will be sued to determine a new correlations' 95/95 limit with the "pooled" database.*

RESPONSE QUESTION 3(a)

When a new fuel assembly design is developed, the geometry is evaluated consistent with the NRC guidelines in the Safety Evaluation Report on the WRB-1 CHF correlation. The geometric parameters critiqued are the fuel rod diameter, rod pitch, heated length, gridded to ungridded cell flow areas, grid spacing, and [ ]+ [+(a,c)
]+ Geo- +(a,c)
metric correlation parameters are a function of the above geometric parameters and thus are not independent. Each geometric parameter should be bounded by the range for grids that have been previously DNB tested and directly support the correlation in question.

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If any one of the geometric parameters does not fall within the respective range, justification must be developed to support the new design. This justification [

]+ CHF tests +(a,c)

would be required. [ ]+ +(a,c)

An example of where a new design was determined to be bracketed by the test data and correlation parameters is the 17x17 VANTAGE 5 Hybrid design. As discussed in Section 4.2.2 of Addendum 2 of WCAP-10444-P-A, this design meets the above mentioned criteria.

At that time, an example of where a new design was not bracketed by the test data or correlation parameters is the 17x17 OFA design. Further elaboration is provided as a response to part (b) of this question.

RESPONSE QUESTION 3(b)

The statistical tests performed would [

]+ +(a,b,c)

The 17x17 OFA example is documented in WCAP-9401-P-A.

The test points [

]+ can be made. As the data [+(a,b,c)

]+ +(a,b,c)

The methods used to determine the pooled database limit DNBR are [

]+ +(a,c)

Question 4. *Three alternative methods are provided in the topical report for evaluating fretting wear depth. Please discuss the criteria that will be applied to determine which of the three alternative methods will be used for evaluating fretting wear. For example, it is anticipated that the second method will be acceptable only if the design being considered and the results of out-of-reactor tests meet particular criteria in relation to previous designs and their out-of-reactor results. Please provide examples. The third method of calculating fretting wear depth includes the use of semi-empirical models. Please provide a description and justification (including comparisons of the empirical model to actual in-reactor wear data up to the burnup limit requested) of why and when these semi-empirical models are justified. Also provide an example of when this third method would be used.*

RESPONSE QUESTION 4

Past experience with both in-reactor operation and out-of-reactor tests indicate that the most important considerations for successful fretting wear performance are the [ ]+ grid +(a,c)
forces. The [ ]+ grid forces must exceed the fretting threshold force at end-of-life (EOL) that +(a,c)
have been determined from in-reactor experience and out-of-reactor tests. The EOL forces are predicted based on in-pile creep and relaxation data.

Other significant considerations are the hydraulic conditions (e.g. crossflow generated by pressure drop mismatches) and grid rod support conditions (e.g distance between support points, spring stiffness).

Regarding the determination of the appropriate method for assuring fretting wear performance, three alternative methods were identified in WCAP-12488:

[

]+ +(a,c)

Use of method #3 is dependent on appropriate [ ]+ +(a,c)
Changes in grid design and/or materials can affect [
]+ In the mid-1970's models such as reported in WCAP-8646 were developed +(a,c)
for early Inconel grid designs. More recently, however, these types of models [have not been used for direct]+ fretting wear evaluation. Where new evaluations [+(a,c)

]+ +(a,c)

The need for [ ]+ is dependent on how the key considerations identified earlier are affected. +(a,c)
For example, if the [ ]+ grid forces are within the fretting experience limits and the material +(a,c)
properties are known, then any other changes (e.g. hydraulic conditions or support conditions) outside current experience bounds [ ]+ For example, if +(a,c)
increased pressure drop differences [

]+ Similarly, [+(a,c)
]+ An example of a +(a,c)
design that utilized [ ]+ was VANTAGE 5H (WCAP-10444, Addendum 2A). +(a,c)

[

]+ +(a,b,c)

[

]+ Examples include Optimized Fuel (OFA) (WCAPs 9500/9400) and +(a,c)
VANTAGE 5 (WCAP-10444).

CATEGORY 3

Question 1. *The recently observed reduction in Zircaloy-4 cladding ductility to values between 0.03 and 0.11% at local burnup levels between 55 and 63 MWd/kgM has raised the concern the mechanical limits imposed to prevent fuel failures during normal and anticipated operational occurrences (AOOs) are non-conservative at and, particularly, above these burnup levels. The cladding mechanical limits of particular concern are a) the uniform elastic plus plastic strain limit of 1.0%, b) the 0.2% irradiated yield strength, and c) the O'Donnel and Langer fatigue curves for irradiated cladding. In addition, no ramp power testing has been performed on prototypic commercial fuel rods to determine failure thresholds at burnup levels greater than ~48 GWd/MTM. Please provide data and analyses that demonstrate that these mechanical design limits remain conservative up to the burnup level requested for Zr-4 cladding or provide alternative mechanical limits that can be defended as being conservative at these higher burnup levels. Also, please provide bounding analyses that demonstrate how much margin exists between calculated values of strain and strain fatigue and their respective design limits up to the requested extension in burnup. How much have these margins decreased as a result of this extension in burnup?*

Question 2. *It is not apparent that the cladding temperature limits proposed by Westinghouse will exclude excessive fuel cladding oxidation at the extended burnup level requested. The NRC Standard Review Plan (SRP) recommends that the effects of cladding oxidation be included in thermal and mechanical analyses. Please discuss how Westinghouse will include the effects of cladding oxidation in their thermal analyses. In the past, Westinghouse has reduced cladding thickness by a proprietary amount to account for cladding imperfections, fretting wear and waterside corrosion. There is a concern that some Westinghouse plants with high primary coolant temperatures and elevated lithium may exceed the Westinghouse reduction in cladding thickness for mechanical analyses and the oxidation thickness used for thermal analyses at the extended burnup levels requested. This is of particular concern since waterside corrosion has been shown to accelerate at extended burnup levels. Please provide the relevant in-reactor waterside corrosion data and how they are applied to thermal and mechanical analyses that demonstrate that Westinghouse conservatively accounts for the effects of Zircaloy-4 waterside corrosion at the burnup limit requested. Also, because cladding waterside corrosion is reactor specific, how does Westinghouse intend to evaluate this phenomenon for those plants with a past history of high Zircaloy corrosion either due to poor controls on water chemistry, high coolant temperatures or elevated lithium levels?*

Question 3. *Does Westinghouse have more recent rod and assembly growth data up to the burnup limit requested? If so, please demonstrate that based on this new data the axial growth models will remain conservative up to the burnup requested.*

RESPONSE QUESTIONS 1,2,3

Since the time that WCAP-12488 was prepared for submittal to the NRC, Westinghouse has completed the development of the VANTAGE+ fuel design, which includes the use of ZIRLO™ material for the fuel rod cladding and fuel assembly guide thimble tubes. VANTAGE+ fuel is explicitly designed for high burnup applications beyond the lead rod target burnup established for Westinghouse fuel in WCAP-10125-P-A. The VANTAGE+ fuel design, including increased fuel discharge burnup, has been submitted for NRC review in WCAP-12610. At this time, it is anticipated that all fuel burnup extensions beyond the target level established in WCAP-10125-P-A will employ the VANTAGE+ design. Therefore, the Extended Burnup section in Section 6.0 of WCAP-12488 is no longer relevant, since licensing issues associated with extended burnup are being addressed as part of the NRC review of the VANTAGE+ design in WCAP-12610. Accordingly, no responses will be provided to Category 3, Questions 1, 2 and 3. Should it become desirable in the future to increase the discharge burnup for current fuel designs beyond the established target limit, a burnup specific report will be submitted to the NRC for review and approval.

SECTION Q

Westinghouse Electric Corporation

Nuclear Manufacturing Divisions

Box 355
Pittsburgh Pennsylvania 15230-0355

July 1, 1994
NTD-NRC-94-4186

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Attention: L. E. Phillips, Acting Chief
Reactor Systems Branch
Division of Systems Technology

Subject: WRB-1 Correlation Applicability, Revision 1

Reference (1): Letter from N. J. Liparulo (Westinghouse) to T. E. Collins (NRC), "WRB-1 Correlation Applicability," NTD-NRC-94-4137, May 20, 1994.

Dear Mr. Phillips:

During a meeting with your Staff concerning our topical report, WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process," on April 21, 1994, in Rockville, the applicability of the WRB-1 DNB correlation to the 17x17 Standard R-grid, Optimized Fuel Assembly (OFA) and VANTAGE 5H grid designs was discussed. Attached is supplemental background information that Westinghouse agreed to supply to the NRC at this meeting. This information clearly demonstrates that the WRB-1 correlation is applicable to the above grid designs. Where appropriate the generic topical and NRC SER are noted to substantiate the use of the WRB-1 DNB correlation for the various Westinghouse grid designs.

This submittal supercedes the information presented in Reference (1). The revised highlighted text presents a summary of the results of the most recent AP600 DNB testing.

This submittal contains Westinghouse proprietary information of trade secrets, commercial or financial information which we consider privileged or confidential pursuant to 10 CFR 9.5(4). Therefore, it is requested that the Westinghouse proprietary information attached hereto be handled on a confidential basis and be withheld from public disclosure.

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Correspondence with respect to any Application for Withholding should reference AW-94-656 and should be addressed to N. J. Liparulo, Manager of Nuclear Safety Regulatory and Licensing Activities, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

H. Sepp for

Nicholas J. Liparulo, Manager
Nuclear Safety Regulatory and Licensing Activities

cc: L. W. Barnett, NRC (MS 12H5)
S. L. Wu, NRC (MS 8E23)
H. Richings, NRC (MS 8E23)

/ssh

Westinghouse Electric Corporation

Nuclear Manufacturing Divisions

Box 355
Pittsburgh Pennsylvania 15230-0355

July 1, 1994
AW-94-656

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Attention: Mr. L. E. Phillips, Reactor Systems Branch Acting Chief, Division of Engineering and Systems Technology

Reference: Letter from N. J. Liparulo (Westinghouse) to L. E. Phillips (NRC), NTD-NRC-94-4186, July 1, 1994

Subject: WRB-1 DNB Correlation Applicability, Revision 1

Dear Mr. Phillips:

The above referenced letter contains information proprietary to the Westinghouse Electric Corporation.

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

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Very truly yours,

Nicholas J. Liparulo, Manager
Nuclear Safety Regulatory and Licensing Activities

cc: Kevin Bohrer / NRC (12H5)

(a,c

Grid Geometries - see attached Figure

Standard Grid - WCAP-8762 - submitted 1976, NRC SER 1978. For the 17x17 configuration, the R-grid (or standard grid) was utilized in testing; the rod diameter is 0.374 inches. At this time, OFA had not been developed.

OFA Grid - WCAP-9401 - submitted 1979, NRC SER 1981. This report contained the description/ results. The pressure drop for this grid is considerably greater than the Standard or V5H grid. WRB-1 was shown to be applicable to OFA (and also still applicable to Standard). SER states (page B-22)

"4.0 Regulatory Position

We have reviewed WCAP-9401 and additional supporting material ... As a result of our review, we conclude the following:

1. The WRB-1 CHF correlation is an acceptable correlation for use in the thermal-hydraulic analysis of OFA.

2. The DNBR limit of 1.17 is an acceptable thermal design limit for the OFA.

..."

VANTAGE 5H Grid - DNB testing of the V5H grid design has been performed as part of the AP600 project. [

]+ +(a,b,c)

The analysis of the IFM data showed that [

]+ +(a,b,c)

SECTION R

Westinghouse
Electric Corporation

Nuclear Manufacturing
Divisions

Box 355
Pittsburgh Pennsylvania 15230-0355

August 29, 1994
NTD-NRC-94-4275

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Attention: R. C. Jones, Chief
Reactor Systems Branch
Division of Systems Technology

Subject: Westinghouse Interpretation of Staff's Position on Extended Burnup [Proprietary]

Reference (1): Letter from M. J. Virgilio (NRC) to M. J. Liparulo)Westinghouse), "Acceptance for Referencing of Topical Report WCAP-12488," Westinghouse Fuel Criteria Evaluation Process" (TAC No. M77257), July 27, 1994.

(2): Davidson, S. L. (Editor), "Westinghouse Fuel Criteria Evaluation Process," WCAP-12488, April 1990.

(3): Letter from N. J. Liparulo (Westinghouse) to R. C. Jones (NRC), "Final Responses to Additional Information on WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process," ET-NRC-93-3819, February 8, 1993.

Dear Mr. Jones:

Westinghouse has reviewed the NRC SER, Reference 1, on the Fuel Criteria Evaluation Process (FCEP), Reference 2. It is stated in the SER conclusions that, " The staff has reviewed the Westinghouse fuel design criteria process described in WCAP-12488 and finds it acceptable for licensing applications up to 60,000 MWD/MTU rod average exposure".

The attachment to this letter notifies the NRC of Westinghouse's interpretation of the above staff position.

This submittal contains Westinghouse proprietary information of trade secrets, commercial or financial information which we consider privileged or confidential pursuant to 10 CFR 9.5(4). Therefore, it is requested that the Westinghouse proprietary information attached hereto be handled on a confidential basis and be withheld from public disclosure.

This material is for your internal use only and may be used solely for the purpose for which it is submitted. It should not be otherwise used, disclosed, duplicated, or disseminated, in whole or in part, to any other person or organization outside the Office of Nuclear Reactor Regulation without the expressed prior written approval of Westinghouse.

Correspondence with respect to any Application for Withholding should reference AW-94-716 and should be addressed to N. J. Liparulo, Manager of Nuclear Safety Regulatory and Licensing Activities, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

Nicholas J. Liparulo, Manager
Nuclear Safety Regulatory and Licensing Activities

cc: L. W. Barnett, NRC (MS 12H5)
L. E. Phillips (MS 8E25)

/ssh

WCAP-12488 was originally submitted to the NRC in April 1990. After discussions with the NRC, Westinghouse submitted a new Section 7.0 and subsequent revisions entitled, " Fuel Performance and Material Properties Models ", Reference 3. Item (5) in Section 7.0 states, as a condition for extending the range of applicability of design and material properties models, that: " The Fuel Criteria Evaluation Process will not be used to extend the range of applicability of any fuel performance model or material property model to burnups greater than [ ]^{a,c} MWD/MTU beyond the target lead rod average licensed burnup limits approved for Westinghouse fuel by the NRC ". Since this documentation is part of the NRC review and approval process, Westinghouse therefore concludes that the FCEP process can be used to justify additional fuel rod average exposures of up to [ ]^{a,c} MWD/MTU beyond the NRC licensed 60,000 MWD/MTU limit. +(a,c) +(a,c)

Based on the above discussion, Westinghouse plans to apply this additional [ ]^{a,c} MWD/MTU extension to the NRC licensed 60,000 MWD/MTU burnup limit, provided that all the fuel design criteria stated in the FCEP continue to be satisfied. +(a,c)

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Attention: Mr. R. C. Jones, Chief, Reactor Systems Branch, Division of Systems Technology

Reference: Letter from N. J. Liparulo (Westinghouse) to R. C. Jones (NRC), NTD-NRC-94-4275, August 29, 1994

Subject: Interpretation of Staff's Position on Extended Burnup [Proprietary]

Dear Mr. Jones:

The above referenced letter contains information proprietary to the Westinghouse Electric Corporation.

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

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

Correspondence with respect to the proprietary aspects of this submittal should reference AW-94-716 and should be addressed to the undersigned.

Very truly yours,

Nicholas J. Liparulo, Manager
Nuclear Safety Regulatory and Licensing Activities

cc: Kevin Bohrer / NRC (12H5)