BAW-10183-A
JULY 1995

FUEL ROD GAS PRESSURE

CRITERION

(FRGPC)

by

D. A. Wesley, D. A. Farnsworth, and G. A. Meyer

B&W Fuel Company

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

February 22, 1994

Mr. J. H. Taylor, Manager
Licensing Services
B&W Nuclear Technologies
3315 Old Forest Road
P.O. Box 10935
Lynchburg, VA 24506

Dear Mr. Taylor:

SUBJECT: ACCEPTANCE FOR REFERENCING OF TOPICAL REPORT BAW-10183P, "FUEL ROD GAS PRESSURE CRITERION" (TAC NO. M82189)

We have reviewed the subject topical report of July 1991, and your responses of July 23 and August 18, 1993, to our requests for additional information. On the basis of our review, we conclude that BAW-10183P provides an acceptable basis for the fuel rod gas pressure criterion 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, B&W 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Should our acceptance criteria or regulations change so that our conclusions as to the acceptability of the report are no longer valid, applicants referencing this topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued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,

Ashok C. Thadani, Director
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

Enclosure:
BAW-10183P Evaluation

ENCLOSURE

SAFETY EVALUATION OF B&W FUEL COMPANY

TOPICAL REPORT BAW-10183P

"FUEL ROD GAS PRESSURE CRITERION"

1.0 INTRODUCTION

In a letter of August 29, 1991, from J. H. Taylor, B&W Fuel Company (BWFC), to the U.S. Nuclear Regulatory Commission (NRC), BWFC submitted a Topical Report BAW-10183P, "Fuel Rod Gas Pressure Criterion," for NRC review.

BAW-10183P describes a fuel rod gas pressure criterion that BWFC intends to apply to existing fuel designs that will allow the rod pressure to exceed the system pressure under certain conditions. This criterion will be applied to future reload applications. This approach, a fuel rod pressure criterion, is consistent with the staff position on 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 our consultant's technical evaluation report (TER), which is attached, as described in this safety evaluation report.

2.0 EVALUATION

The attached TER provides the evaluation.

3.0 CONCLUSIONS

The staff has reviewed the BWFC fuel rod gas pressure criterion described in BAW-10183P, and finds it acceptable for licensing applications.

In addition, the staff also determines that (1) if LOCA LHGR's become limiting at extended burnup levels for any BWFC design applications, the LOCA LHGR analyses should be submitted for NRC review, and (2) if there are changes involving the power peaking maps or other input parameters for the response surface applications, the core protection analysis should be submitted for review.

TECHNICAL EVALUATION REPORT OF THE TOPICAL REPORT BAW-10183P, ENTITLED "FUEL ROD GAS PRESSURE CRITERION (FRGPC)"

D. D. Lanning
C. E. Beyer

October 1993

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

Pacific Northwest Laboratory
Richland, Washington 99352

ABBREVIATIONS

BOL	Beginning-of-Life
BWFC	Babcock & Wilcox Fuel Company
DNB	Departure from Nucleate Boiling
EOL	End-of-Life
FGR	Fission Gas Release
FRGPC	Fuel Rod Gas Pressure Criterion
FIN	Financial Identification Number
LHGR	Linear Heat Generation Rate
LOCA	Loss-of-Coolant Accident
NRC	U.S. Nuclear Regulatory Commission
PNL	Pacific Northwest Laboratory
RCS	Reactor Coolant System
RSM	Response Surface Method
TER	Technical Evaluation Report

CONTENTS

1.0 INTRODUCTION

This Technical Evaluation Report (TER), prepared for the U.S. Nuclear Regulatory Commission (NRC) by Pacific Northwest Laboratory (PNL)[1] under Financial Identification Number (FIN) I2009, is a review of the methodology by which Babcock and Wilcox Fuel Company (BWFC) will apply a fuel rod gas pressure criterion (FRGPC) in limiting the calculated rod pressures for licensing of BWFC-fuelled reactors. This methodology is described in the topical report BAW-10183P (Reference 1) and was amplified by BWFC's letter of July 23, 1993 (Reference 2) in response to NRC's request for additional information (Reference 3). A subsequent submittal from BWFC (Reference 4) further modified the proposed methodology.

The current criterion limiting fuel rod internal pressure is, that this pressure shall not exceed the reactor coolant system (RCS) pressure during normal operation. BWFC is seeking to revise this criterion to the following:

"The internal pressure of the peak fuel rod in the reactor will be limited to a value below that which would cause (1) the fuel-clad gap to increase due to cladding outward creep during steady-state operation and (2) extensive departure from nucleate boiling (DNB) propagation to occur. [In no case will the internal pressure of the peak fuel rod exceed the RCS pressure by more than a proprietary value during normal operation.]"

The fuel-cladding gap is judged to increase if, at any axial node for any time step in the peak rod power history, the cladding outward creep rate exceeds the fuel pellet swelling rate (with the exception of nodes with very low linear heat generation rates). BWFC has demonstrated that gap size increases over time frames of interest for axial nodes with continued linear heat generation rates (LHGRs) of less than a small proprietary value are inconsequential.

The derivation of the swelling rate as a function of fission rate, operating time and material (densification) parameters is presented in the topical report, together with the derivation of the cladding outward creep rate as a function of fast neutron flux, cladding hoop stress, cladding temperature, and operating time. These derivations are reviewed in Section 2.

A related issue, due to the possible occurrence of cladding tensile stresses under the revised criterion, is the extent of hydride reorientation and consequent degradation of cladding strength and ductility. BWFC presents in-reactor and ex-reactor test data in the topical report to confirm that hydride reorientation effects are not significant for the projected maximum conditions of cladding stress, temperature, and hydride content. These data are reviewed in Section 3.

[1] Operated for the U.S. Department of Energy by Battelle Memorial Institute under contract DE-AC06-76RLO 1830.

Another related issue is the potential for the additional rod pressure in a limited number of rods to cause cladding ballooning and DNB propagation during projected loss-of-coolant accidents LOCAs). BWFC presents extensive statistical analysis of core-wide fuel rod behavior and develops a core-protection criterion that is asserted to result in no DNB propagation. These analyses are evaluated in Section 4.

BWFC also routinely checks for any change in LOCA analysis limitations on the LHGRs for any change to fuel performance analysis methodology. Their discussion of this relative to the FRGPC is reviewed in Section 5.

Finally, BWFC presents the methodology by which the FRGPC and the core protection criterion will be applied in licensing submittals. This methodology is reviewed in Section 6. PNL conclusions regarding the use of the FRGPC are given in Section 7.

2.0 THE GAS PRESSURE CRITERION

The gas pressure criterion evaluation depends on the simultaneous evaluation of local, instantaneous values for the fuel pellet diametral strain rate due to swelling and the corresponding cladding outward diametral creep rate. BWFC's derivation of these two rates as a function of operating condition and material parameters is reviewed in this section. It is concluded that cladding creep rates and the pellet diametral strain rates are appropriately bounded by the conservatisms in the derivation and application for the current BWFC fuel cladding type.

However, a secondary pressure criterion is proposed by BWFC: the maximum internal rod pressure will be limited to less than a proprietary value above RCS pressure. This is proposed to preclude significant consequence of gap size changes. BWFC has demonstrated, and PNL has confirmed, that this secondary criterion does provide an appropriate margin against gap size increases which would lead to thermal feedback to fuel temperatures and fission gas release rate, and hence a runaway cladding stress/strain and failure in normal operation.

2.1 FUEL PELLET SWELLING RATE

The fuel pellet densification and swelling model used by BWFC in the TACO3 code is based on a model by Marlowe (Reference 5) for pellet density as a function of fission rate swelling rate, and material (densification) parameters. This model includes the effect of porosity reduction due to in-reactor densification, and the effect of partial swelling into the porosity. A standard value, previously accepted by NRC in its review of the TACO3 code, is used for the solid-fission product isotropic swelling rate. No account or credit is taken for fission-gas induced pellet swelling, which may occur to a significant degree in the high-burnup, unconstrained conditions in which the FRGPC will be applied. This is a further conservatism in the derivation.

To derive the pellet diametral strain rate due to swelling, BWFC first differentiated the density function with respect to time, and then translated the resulting density change rate to diametral strain rate, assuming isotropic volumetric changes. The result is a diametral strain rate that is equal to the fission rate times the swelling rate (adjusted for isotropic swelling), times a factor that is dependent on burnup and the densification parameters.

BWFC made one further alteration, which introduced considerable conservatism in the calculated strain rates. Detailed postirradiation examinations on companion rods were carried out (under the BWFC extended burnup program) following 3, 4, and 5 cycles of irradiation. Based on a comparison between fuel stack changes and rod diametral changes between the 4-cycle rods and the 5-cycle rods, BWFC concluded that the swelling under "hard-contact" (cladding-constrained) conditions is anisotropic, with the diametral strain rate being significantly less than that derived from the isotropic strain rate. Hence the isotropic strain rate is reduced by a proprietary factor in the BWFC equation for pellet diametral strain rate due to swelling (Equation 2.6 in Reference 1). This may be quite conservative for the application of evaluating gap size changes in rods where the internal gas pressure exceeds the RCS pressure because, in those cases, the cladding no longer compresses the fuel, and the differential swelling probably returns to being isotropic.

PNL concludes that the BWFC model for fuel pellet diametral strain rate is conservative, due both to the choice of the volumetric swelling rate and the anisotropic factor applied to the derived diametral strain rate.

2.2 CLADDING OUTWARD CREEP RATE

BWFC has derived an expression for cladding outward creep strain rate at high burnup by differentiating a high-burnup creep strain model with respect to time. The creep strain model was developed by statistical parameter fitting of a standard-form creep strain model to selected high-burnup (high-fluence) data from fuelled and unfuelled rods with basically one Zircaloy-4 cladding type. The creep strain data base consisted entirely of compressive (inward) creep data. To assure that the derived model represented or bounded outward creep behavior, the model was tested against outward creep data from Canadian sources on Zircaloy-2 materials. The results indicated that the model would bound outward creep behavior of Zircaloy tubing.

In response to NRC requests, BWFC presented residuals plots for the model versus the 833-point data base as a function of temperature, stress, operating time, and fast neutron flux. These plots demonstrated that there was no statistical bias of the model relative to the primary parameters for the given data base (especially relative to temperature), and that the database ranges on the parameters span the ranges of interest.

A 95/95 upper tolerance bound was derived for the model relative to the creep strain data using standard statistical techniques for normally distributed variables; and this was translated to a slightly larger bound on the creep strain rate.

One feature of the creep strain expression presented in BAW-10183P that has been reviewed at length is the temperature dependence in the cladding creep model. The temperature dependence of the BWFC model is different from that concluded by several other investigations of Zircaloy creep (References 6 and 7). BWFC contends that, at high burnup, irradiation hardening changes the dependence on thermally-activated creep, and that therefore, at least for the current BWFC Zircaloy-4 cladding type, the temperature dependence of the creep strain rate over the operating temperature range of interest (500 to 700 °F) is modeled correctly based on the BWFC creep data. The BWFC creep strain rate model appears to be satisfactory for the BWFC cladding type over the temperature range of interest.

In their response to NRC questions, BWFC also considered the cladding creep response to sudden stress reversals, as might occur in an overpower transient accompanied by significant fission gas release that increases the rod gas pressure above the RCS pressure and rapidly reverses the cladding hoop stress from compressive to tensile. In this situation, primary (thermally-driven) creep would reoccur to a limited degree over a limited time. The creep rates in the BWFC creep strain rate model were found to adequately address the reoccurrence of primary creep when cladding stresses change from compressive to tensile (as deduced from ex-reactor expansion tests on irradiated tubing).

The BWFC creep strain rate model was compared to several other open-literature models as a function of fast flux, stress, and temperature (References 6, 7, 8, 9, and 10). As noted above, the BWFC temperature dependence is different from that of any other model considered, and the BWFC predicted creep strain rates exceed those of other models for part of temperature range of interest, i.e., from 500 to approximately 625°F. However, the temperature dependence in the BWFC creep strain rate model appears to be sarisfactory based on the BWFC creep strain data presented for the current BWFC Zircaloy cladding type.

In addition, PNL also finds that, at the tensile hoop stresses corresponding to the BWFC nominated maximum pressure differential, the cladding outward creep rates demonstrate considerable margin relative to predicted pellet diametral strain rates due to isotropic swelling (as opposed to the anisotropic swelling which BWFC conservatively uses), across the full range of expected LHGR and fast neutron flux. For example, for the Mark B design, the margin between the best-estimate cladding creep rate (at the BWFC proprietary limit above RCS pressure) and the isotropic pellet strain rate (which PNL considers the best-estimate function) is a factor of 2 or more except at very low LHGRs. The margin is even greater for the Mark BW rod design.

In view of the above findings, PNL concludes that the FRGPC does have adequate conservatism when applied to the current BWFC Zircaloy-4 cladding type that is represented by its current data base.

2.3 EFFECTS OF CLADDING OXIDATION

BWFC does account for the effect of cladding corrosion in thermal calculations for cladding and fuel. BWFC has stated (Reference 2) that the wall-thinning effects of cladding oxidation are not accounted for in the calculation of cladding stress, which factors into the calculation of cladding creep. The maximum expected oxide layer thickness is, however, about 100 microns which corresponds to about 10% wall thinning. This corresponds to a 10% underestimation of the cladding stress at a given differential pressure or a related 9% underestimation of the cladding creep rate. On the other hand, BWFC uses the thin-wall approximation to calculate cladding hoop stress which results in about 7% overestimation of the wall-average tensile stresses and corresponding creep rates. Because the two effects are compensating, and because the attainment of the liftoff criterion is most closely approached at rod axial locations with low LHGRs (where the cladding oxidation is similarly reduced), PNL concludes that the omission of stress correction for cladding wall thinning is acceptable, given the current BWFC design limits on rod-average burnup and oxide layer thickness.

3.0 CLADDING HYDRIDE REORIENTATION

The reversal of stresses in the cladding when the internal pressure exceeds RCS pressure, leads to the possibility of hydride reorientation from parallel to perpendicular to the cladding surface. This reorientation would significantly reduce the strength and fracture toughness of the cladding. In the topical report, BWFC described the results of uniaxial mechanical strength and ductility measurements on hydrided cladding which indicated no such effects were detectable within the stress and temperature ranges of interest. To promote hydride precipitation in the radial direction, the hydrided samples (with hydrogen concentrations ranging from 0 to 710 ppm), were heated to 750 or 775 °F with a bounding value for tensile stress, and then cooled under stress to the mechanical test temperatures (70 and 400 °F). No measurable loss-of-strength or loss-of-ductility was recorded, indicating that any hydride reorientation was too little to have significant effect.

BWFC presented supplementary information in Reference 2 to indicate that the hydrogen contents represented in these tests bounded the expected levels in BWFC high-exposure cladding. The maximum stress level for the hydride reorientation tests also bounded the tensile hoop stresses that would result from the maximum allowed differential pressure.

BWFC also presented in Reference 2 the results of an in-reactor power ramping experiment in which 1-meter segments from two 5-cycle segmented rods irradiated in a U.S. PWR were power-ramped to high LHGRs in the R2 reactor at Studsvik, Sweden. The rods had circumferential ridges prior to ramping and these ridges increased in height as a result of the ramping. Nevertheless, post-test sectioning and metallography revealed little or no hydride reorientation and no cladding defect occurred. The maximum hydrogen content in these tested segments was not reported, but was probably in the 300 to 400 ppm range based on normal-operation oxidation and hydrogen pickup data quoted by BWFC in Reference 2.

It should be noted that the general hydrogen concentration in high-burnup fuel rod cladding greatly exceeds the solubility for hydrogen in Zircaloy; therefore, hydride precipitates will exist even at normal-operation cladding temperatures. However, BWFC concludes, on the basis of the two test series cited, that the cladding stresses permitted under application of the proposed FRGPC are not sufficient to cause significant hydride reorientation. PNL concurs with this conclusion. Also, note that the duration of the tests cited are short relative to the time periods for which cladding could be held in tensile stress by application of the FRGPC.

4.0 DNB PROPAGATION

The potential for fuel rod failure propagation due to cladding ballooning and DNB propagation was analyzed by BWFC separately for the Mark B and Mark BW rod types, and the results were presented in Reference 1. For the purpose of this analysis, failure was defined to occur if a rod simultaneously had an internal pressure greater than the RCS pressure, and was in DNB due to power peaking. The Response Surface Method (RSM) was applied to find the intersection of two statistical distributions: the distribution of DNB caused by power peaking, and the distribution of rod internal pressures. The analysis was repeated at ascending levels of severity for operating conditions, as described below.

The development of both statistical distributions was carried out with appropriate regard for model and input uncertainties. In the case of the DNB distribution, input uncertainties were handled differently for the two core types. For Mark B rods, inputs were compounded at individually conservative values. For Mark BW rods, the impacts of the inputs were bounded statistically through Monte Carlo sampling. For both core types, the model uncertainties were bounded using pre-established DNBR design limits. In the case of the internal pressure distribution, pre-established uncertainties on inputs and models were used to develop response surfaces, which could then be sampled under various peaking assumptions to establish differing distributions of rod internal pressure.

The occurrence of failure was assessed at four ascending levels of severity for operating conditions and power peaking:

Level 1 - Normal Operation
Level 2 - Extended Power Peaking
Level 3 - Extended Burnup
Level 4 - Extended Burnup and Extended Peaking Simultaneously

The level 4 condition is not realistically achievable because the fuel is sufficiently burned to limit the severity of the transient. However, this case is included to provide an outer bound for analysis.

The results of the RSM analyses were that less than 1 failure per core quadrant per cycle could be predicted, up through Level 3. Thus, BWFC concluded that the application of the FRGPC would result in no additional failed rods, for the purpose of core and radiological analyses. PNL concurs with this conclusion, as it is demonstrated in this submittal for the Mark B and Mark BW rod types. BWFC has stated that they will repeat these analyses for other rod types, using the same methodology.

5.0 IMPACT ON LOCA

BWFC routinely calculates the limiting peak LHGR's that can be permitted relative to the predicted results of hypothetical loss-of-coolant accidents. BWFC has provided an example of the methodology that would used to determine the impact of the revised rod pressure limit. As a result, BWFC has shifted the LHGR limit as a function of burnup for the B&W 177-FA lowered-loop plants, operating at 2772 MWt, as noted in Figure 1 of Reference 1. Based on audit calculations, PNL concurs with the magnitude of this LHGR shift, and with the indicated continued reduction of peak LHGR with increasing burnup, if the limiting criterion is understood to be rod internal pressure during normal operation, not peak cladding temperature or pressures attained in the LOCA analysis.

BWFC has committed to performing similar analyses using the same methods on a plant-cycle specific bases during reload analyses for specific fuel designs, to determine if LOCA LHGR limits at extended burnup need to be changed.

From these analyses it appears that limiting LHGRs for LOCA will not be limiting for normal-operation LHGRs for BWFC designs at extended burnups. If LOCA LHGRs become limiting at extended burnup levels for any BWFC design applications, these should be submitted to NRC for review.

PNL concurs with the methodology utilized by BWFC for determining the shift in limiting LHGRs versus burnup for LOCA.

6.0 APPLICATION METHODOLOGY

BWFC has outlined their methodology for application of the FRGPC in Reference 1. The applications with regard to fuel internal pressure and DNB propagation are reviewed separately below.

6.1 FUEL ROD PRESSURE CRITERION

BWFC proposes to use Equation 2.6 from Reference 1 to calculate the fuel pellet diametral strain rate due to fuel swelling using the modifications noted in Reference 4. The cladding creep strain will be calculated using Equation 2.8 from Reference 1, times a factor that provides a 95/95 tolerance interval based on the creep data. Cladding liftoff will be judged to occur if this upper-bound cladding creep rate exceeds the pellet diametral strain rate due to swelling. The liftoff will be discounted for those nodes for which the ongoing LHGR is less than a small proprietary value, because, as BWFC demonstrates in Reference 4, the resulting fuel temperature increases are inconsequential for LHGRs less than this value. The rod internal pressure will be limited to the proprietary value above RCS pressure, or the pressure required to achieve cladding liftoff, whichever is smaller.

PNL analyzed the BWFC proposed liftoff criterion as applied to both the Mark B and Mark BW rod designs, at the proprietary differential pressure above RCS pressure. It was concluded that, 1) although liftoff may occur at very low LHGRs the consequences in terms of gap size increase and fuel temperature increase over time frames of interest are negligible and inconsequential; and 2) there is adequate conservatism in the application of the models for cladding creep and pellet swelling rates, upon which the liftoff criterion depends.

Based on the above analysis, PNL concludes that the methodology proposed for applying the lift-off criterion is acceptable.

6.2 DNB PROPAGATION

BWFC states that it will re-perform its core protection evaluation (via the response surface method) against the proposed core protection criterion (99.99%) if the currently-used power peaking maps change. PNL concurs, and adds that if any of the other crucial assumptions or statistical distributions that form the input for the response surface application change, then the core protection evaluation should be re-performed and resubmitted to NRC.

7.0 CONCLUSIONS

The FRGPC represents an alternate method by which BWFC will limit the peak LHGRs for fuel rods as a function of burnup. Under this alternate method, the rod internal pressure for a limited number of rods will be permitted to exceed the RCS pressure, but will not exceed the smaller of the following: the proprietary limit above RCS pressure, or that pressure which would cause cladding liftoff to occur at significant LHGRs. The number of rods which will exceed the RCS pressure is limited by a core protection criterion which declares that the number of such rods, which also experience DNB in overpower transients, shall not exceed 0.01% of the rods in the core.

Upon reviewing the proposed criteria and their application, PNL concludes that calculation of fuel pellet diametral strain rate due to swelling is conservative and that the calculation of cladding creep strain rate is likewise conservative, for the current BWFC Zircaloy-4 rod cladding type, as represented by its current data base.

BWFC has also stated that the core protection and LOCA analyses will be repeated for rod types other than the current Mark B and Mark BW types presented in the submittal.

PNL concludes that the FRGPC is acceptable for licensing applications for BWFC fuel rod designs.

8.0 REFERENCES

1. D. A. Wesley, D. A. Farnsworth, and G. A. Meyer July, 1991. "Fuel Rod Gas Pressure Criterion", BAW-10183P Babcock and Wilcox Company, Lynchburg, VA.

2. Letter, J. H. Taylor (BWFC) to R. C. Jones (NRC/NRR). July 23, 1993. "Response to Request for Additional Information on Topical Report BAW-10183P".

3. Letter, R. C. Jones (NRC/NRR) to J. H. Taylor (BWFC). June 30, 1993. "Request for Additional Information on Topical Report BAW 10183P".

4. Letter, J. H. Taylor (BWFC) to R. C. Jones (NRC). August 18, 1993. "Computational Error in BAW-10183P, Fuel Rod Gas Pressure Criterion"

5. M. O. Marlowe. July 1973. "In-Reactor Densification Behavior of UO_2". NEDO-122440, General Electric Co., San Jose, CA.

6. V.N. Shah, and J.E. Tolli. 1987. "A Regression Model for Zircaloy Cladding in-Reactor Creepdown: Database, Development, and Assessment." in <u>Nuclear Engineering and Design</u> vol 101, pp 233 to 247.

7. V. Fidleris. 1988. "The Irradiation Creep and Growth Phenomena" in <u>Journal of Nuclear Materials</u>, vol 159, pp 22-42.

8. P. A. Ross-Ross and C. E. L. Hunt. 1968. "The In-Reactor Creep of Cold-Worked Zircaloy-2 and Zirconium 2.5 wt% Niobium Pressure Tubes". <u>Journal of Nuclear Materials</u>, vol. 26.

9. J. H. Gittus, et al. 1980. Theoretical Analysis of Cladding Stresses and Strains Produced by Expansion of Cracked Fuel Pellets" <u>Nucl. Appl. Tech.</u>, vol.9.

10. D. G. Franklin et al. 1983. "Creep of Zirconium Alloys in Nuclear Reactors". ASTM-STP-815.

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

September 9, 1994

Mr. J. H. Taylor, Manager
BWNT Licensing Services
B&W Nuclear Technologies
3315 Old Forest Road
P. O. Box 10935
Lynchburg, VA 24506-0935

Dear Mr. Taylor:

Reference: BAW-10183P, "Fuel Rod Gas Pressure Criterion," July 1991.

Pursuant to your letter dated June 2, 1994, you requested modifications to our Safety Evaluation Report (SER) and the accompanying Technical Evaluation Report (TER) of the referenced topical report. The first three modifications are editorial corrections to the TER. The fourth modification to the TER and the associated modification in the SER are intended to clarify BWFC methodology. Our consultant PNL and the staff have reviewed the modifications and concur with the proposed revisions.

The revised SER is enclosed per your request.

Gary M. Holahan, Director
Division of Systems Safety Analysis
Office of Nuclear Reactor Regulation

Enclosures:
As stated

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

ENCLOSURE

SAFETY EVALUATION OF B&W FUEL COMPANY
TOPICAL REPORT BAW-10183P
"FUEL ROD GAS PRESSURE CRITERION"

1.0 INTRODUCTION

In a letter of August 29, 1991, from J. H. Taylor, B&W Fuel Company (BWFC), to the U.S. Nuclear Regulatory Commission (NRC), BWFC submitted a Topical Report BAW-10183P, "Fuel Rod Gas Pressure Criterion," for NRC review.

BAW-10183P describes a fuel rod gas pressure criterion that BWFC intends to apply to existing fuel designs that will allow the rod pressure to exceed the system pressure under certain conditions. This criterion will be applied to future reload applications. This approach, a fuel rod pressure criterion, is consistent with the staff position on 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 our consultant's technical evaluation report (TER), which is attached, with the following revisions to the TER.

2.0 REVISIONS TO TER

TER page 1.2: in the first sentence change ..."loss of coolant accidents (LOCAs)" to..." normal operation and anticipated operational occurrences," TER page 2.3, second paragraph, last sentence - satisfactory is spelled incorrectly,

TER page 3.1, third paragraph, change "5-cycle" to "4-cycle,"

TER page 6.1, section 6.2, change the first paragraph as follows:

BWFC states that it will re-perform its core protection evaluation (via the response surface method) against the proposed core protection criterion (99.99%) if the critical input parameters, as defined in Tables 4.11 and 4.12 of reference 1, change. PNL concurs, and agrees that, if any of the other crucial assumptions or statistical distributions that form the input for the response surface application change, then the core protection evaluation should be re-performed using the methods defined in reference 1.

3.0 EVALUATION

Based on our consultant PNL recommendations and the staff review of the TER, we agree with the PNL evaluation and conclude that the TER provides adequate technical basis to approve BAW-10183P.

4.0 CONCLUSIONS

The staff has reviewed the BWFC fuel rod gas pressure criterion described in BAW-10183P, and finds it acceptable for licensing applications.

In addition, the staff also determines that (1) if LOCA LHGR's become limiting at extended burnup levels for any BWFC design applications, the LOCA LHGR analyses should be submitted for NRC review, and (2) if there are changes involving the critical input parameters for the response surface applications, the core protection analysis shall be revised according to the method described in BAW-10183P.

Abstract

This report describes the B&W Fuel Company revised fuel rod gas pressure criterion. The new criterion allows a small percentage of the rods within the core to operate with a gas pressure that exceeds the nominal reactor coolant pressure. This criterion was adopted to provide a more realistic basis for ensuring that the fuel rod heat transfer characteristics and cladding integrity are maintained during operation while increasing fuel cycle design flexibility. The safety implications of the revised criterion were investigated to assess the effect on fuel-cladding gap stability, hydride reorientation, DNB propagation, and LOCA limits.

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1.0 INTRODUCTION

The current criterion[1,2] used for licensing B&W fueled reactors limits the fuel rod internal gas pressure to less than the nominal reactor coolant system (RCS) pressure. This criterion was chosen as a convenient and conservative basis for ensuring that the cladding mechanical integrity and subchannel flow characteristics were maintained within acceptable limits during operation.

Economics, limited facilities for spent fuel storage, and the efficient utilization of current resources are incentives for extending the burnup of nuclear fuel. The impetus for extending the burnup, and consequently increasing the fuel rod internal gas pressure, caused B&W to reevaluate this criterion. This reevaluation process indicated that the current criterion is unnecessarily conservative and led to the adoption of the following revised criterion:

"The internal pressure of the peak fuel rod in the reactor will be limited to a value below that which would cause (1) the fuel-clad gap to increase due to outward cladding creep during steady-state operation and (2) extensive DNB propagation to occur."

The revised criterion permits a small number of rods in the core to operate with internal gas pressures greater than the nominal RCS pressure. Operation with fuel rods that have internal pressures greater than the RCS pressure has always been available within the NRC Standard Review Plan guidelines as long as justification could be provided. The justification for using the revised criterion is based on the hypothesis that preventing the fuel-clad gap from opening late in life will preserve the fuel-clad heat transfer characteristics and prevent higher fuel temperatures that may increase fission gas release. It should be noted, however, that the current or revised criterion does not require the internal pressure to remain below the transient RCS pressure during accident conditions. During accidents such as Accidental Depressurization of the RCS, Steamline Break, or LOCA, the transient pressure may fall well below the nominal RCS value. Under these conditions, even with the current overly conservative criterion, some rods would have internal pressures greater than the transient RCS pressure.

It is the purpose of this report to review the impact of the revised internal gas pressure criterion on fuel rod integrity, DNB propagation, and LOCA concerns. It will be shown that the revised criterion does not result in any new safety concerns or the violation of any stated safety criterion for accident events.

2.0 FUEL-CLADDING GAP STABILITY

The objective in establishing a fuel rod internal gas pressure limit is to ensure that fuel rod integrity is preserved during the operation of the reactor. This includes a reasonable assurance that the limit prevents excessive fuel temperatures, excessive internal rod gas pressures due to fission gas release, and excessive cladding stresses and strains. These effects will be held within acceptable bounds provided that the fuel rod heat transfer characteristics have not been degraded by the rod pressure. It has been postulated that a cladding diametral creep-out rate that exceeds the fuel diametral growth rate will open a previously closed gap and lead to higher fuel temperatures. This hypothesis is undoubtedly somewhat conservative because it ignores fuel relocation and fuel-clad bonding effects that have been observed in extended burnup fuel. This hypothesis does, however, provide a conservative and convenient means of establishing an internal pressure limit and will be observed in this report.

2.1 Fuel Diametral Growth Rate

The diameter of a fuel pellet positioned within a fuel rod operating in a reactor is primarily a function of the fuel thermal expansion and the densification and swelling characteristics. Fuel elasticity and creep are often neglected because these effects are comparatively small under typical operating conditions and accurate fuel diameter predictions can be obtained by ignoring these effects. One issue that complicates the determination of fuel pellet diameters is that pellets generally do not remain intact and unbroken within a fuel rod. A broken pellet does not have a clear geometric boundary as a reference. This fact, however does not pose a problem in determining when the cladding diametral creep-out rate exceeds the fuel diametral growth rate at extended burnups. "Hard" fuel-clad contact is normally well established before extended burnup operation begins. Under "hard" fuel-clad contact conditions, the cladding inside surface effectively conforms to the pellet outside cylindrical surface. Therefore, the pellet outside diameter is equal to the cladding inside diameter.

[b,c,d,e]

[b,c,d,e]

[b,c,d,e]

2.2 Cladding Diametral Creep-Out Rate

During the last several decades, an enormous amount of material has been published on the creep of Zircaloy alloys subjected to reactor conditions. Many of the classic articles are referenced in reviews presented on this subject[5,6].

[b,c,d,e]

[b,c,d,e]

[b,c,d,e]

[b,c,d,e]

Tables 2.1 through 2.4
on Pages 2-9 through 2-10

and

Figures 2.1 through 2.3
on Pages 2-11 through 2-13

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3.0 CLADDING HYDRIDE REORIENTATION

Cladding hydride reorientation becomes a potential concern when rod internal pressures exceed the RCS pressure. Hydride reorientation can occur when the cladding of a rod that has been exposed to higher temperatures is cooled with tensile stresses. Thus, cladding is particularly susceptible to hydride reorientation under conditions leading to reductions in both the RCS pressure and temperature. A sufficient degree of hydride reorientation can reduce cladding strength and ductility. [b,c,d,e]

A nominal amount of hydrogen is present in virtually all commercially produced nuclear fuel rod cladding. The hydrogen concentration increases during life due to effects such as hydrogen uptake from waterside corrosion. When the hydrogen concentration reaches the terminal solubility limit, the hydride phase (ZrH_2), begins to precipitate out as platelets. The platelets tend to form along a habit plane nearly parallel to the basal plane. In nuclear fuel rod cladding with its high degree of radial texture, the habit plane is parallel with the cladding surface. This causes the hydrides to form with a circumferential orientation. At higher cladding temperatures, a portion of the hydrogen will return to solution. When the cladding is cooled with a tensile stress of sufficient magnitude, the hydrides will precipitate out in a plane perpendicular to the applied principal tensile stress; a radial orientation. The tendency for hydrides to form in a radial orientation is a function of the cladding fabrication process, the minimum temperature attained during cooldown, and the magnitude of tensile stresses present. Radially oriented hydrides have been shown to have a detrimental affect on cladding strength and ductility. The formation of radially oriented hydrides is of particular concern during anticipated operational events where the system pressure and temperature decrease.

[b,c,d,e]

[b,c,d,e]

[b,c,d,e]

Figures 3.1 through 3.3
on Pages 3-5 through 3-7

are Proprietary

4.0 DNB Propagation

One concern associated with fuel rods operating with an internal gas pressure greater than the RCS pressure is the potential for DNB propagation. A fuel rod operating with its internal pressure above the RCS pressure and also in DNB would have a small probability of ballooning, thus promoting DNB propagation to surrounding rods. However, DNB probability is related to the rod power, or peaking level, which decreases with burnup, while the probability of being above the RCS pressure increases with burnup. The purpose of this section is to demonstrate that DNB propagation can be precluded by implementing criteria to prevent the combined occurrence of DNB and operating above the RCS pressure. [b,c,d,e]

Typical core design peaking and burnups will be examined first. Peaking, burnup, and finally combined peaking and burnup will then be extended to the maximum allowable limits. These extended core conditions will then be analyzed for fuel failure and will establish and/or confirm subsequent peaking and burnup limits.

4.1 Protection Level

[b,c,d,e]

This criterion thus precludes pin failure propagation from a combined DNB-above RCS pressure event.

4.2 Methods

In order to determine a failure rate for a given core, [b,c,d,e].

The analyses of a number of normal and limiting core conditions are then conducted to establish confirmation limits for subsequent core reloads.

4.3 Core Types

The analysis presented here applies to the Mark B and the Mark BW cores. Separate analyses for these cores are required due to the physical differences. The same methodology also can be applied to other core types.

The Mark B core applies to a B&W-designed reactor. This core consists of 177 fuel assemblies, each of which has 208 active fuel rods in a 15 by 15 matrix. The Mark B rod diameter and pitch are 0.430 and 0.568 inches, respectively. The Mark B spacer grids do not have mixing vanes. The rated power of the Mark B core in this analysis is 2772 megawatts thermal.

The Mark BW core in this example applies to a Westinghouse-designed four loop reactor. This core consists of 193 fuel assemblies, each of which has 264 active fuel rods in a 17 by 17 matrix. The Mark BW rod diameter and pitch are 0.374 and 0.496 inches, respectively. The Mark BW spacer grids have mixing vanes. The rated power of the Mark BW cores in this example is 3411 megawatts thermal.

4.4 Rod Pressure Models

Fuel rod internal gas pressure is a function of many independent variables including rod geometry, fuel characteristics, power level, and burnup. [b,c,d,e] The power histories used for the Mark B and Mark BW core analyses are shown in Figure 4.1 . The resulting Mark B and Mark BW pressure RSM's are shown in Figures 4.2 and 4.3, respectively.

4.5 Pressure Uncertainties

Three uncertainties are applicable to the pressure calculation.

[b,c,d,e]

The remaining uncertainties, which include manufacturing, code, and power history, have been extensively developed[1] utilizing Monte Carlo propagation techniques. They are tabulated as a function of burnup in Tables 4.1 and 4.2 for the Mark B and Mark BW fuel, respectively. [b,c,d,e]

4.6 DNBR Models

The rod critical heat flux (CHF) is a function of many independent variables including fuel assembly geometry, flow distributions, power distributions, etc. For this work, however,[b,c,d,e].

4.7 DNBR Uncertainty

There are two types of uncertainties in the calculation of CHF: the correlational uncertainty and the uncertainty in the independent variables. In the current treatment of the Mark B core, [b,c,d,e]

[b,c,d,e]

4.8 Core States for Analysis

All of the above calculational models and individual uncertainties for each core type were combined into computer codes which perform the propagation analysis for each type of fuel. The listings for these codes can be found in Appendix A for the Mark B and Appendix B for the Mark BW. The remaining input to this computer code is the description of the core condition for the current analysis. This consists of the [b,c,d,e].

4.9 Conditions for Analysis

Condition I events are those events expected frequently during the course of normal operation. Condition II events are the anticipated transient events which are presumed to occur with moderate frequency during the life of a plant. Condition III and IV events include design basis accidents and accidents with a very low expected frequency of occurrence.

Fuel failure from DNB propagation during Condition I and Condition II events, as postulated in this analysis, is precluded by the limits imposed on the core itself; no DNB is permitted during these events. Condition III and IV events are either isolated to a minute section of the core or are of a duration and condition that precludes failure propagation due to simultaneous DNB and overpressure operation.

The major danger to a core, from DNB propagation failure, will be when high peaking and high burnup conditions occur simultaneously in a sufficient number of fuel rods to cause failure which is simultaneous DNB and overpressure. While the conditions necessary to produce any significant probability of failure are highly unlikely, creating and analyzing a core under these conditions will establish overall limits of operation sufficient to preclude failure. From this basis, then, [b,c,d,e,].

[b,c,d,e]

4.10 Results

The analysis was performed, as discussed above, at four levels for both types of fuel. The Mark B results are shown in Tables 4.3 through 4.6 [b,c,d,e]. The corresponding Mark BW results are shown in Tables 4.7 through 4.10. [b,c,d,e]

4.11 Summary

[b,c,d,e]

[b,c,d,e]

Any reanalysis performed to evaluate the effects of changes would be performed with the methods outlined in this report.

Tables 4.1 through 4.12
on Pages 4-10 through 4-15

and

Figures 4.1 through 4.7J
on Pages 4-16 through 4-42

are Proprietary

5.0 IMPACT ON LOCA

During a postulated loss-of-coolant-accident (LOCA), when the reactor coolant system (RCS) pressure drops below the fuel rod internal pressure, the cladding may swell and rupture. The reactor thermal and hydrodynamic behavior during a postulated LOCA depends upon the type of accident, the time at which swelling and rupture occur, and the resulting coolant flow blockage. B&W has developed a system of computer codes and related analytical procedures under the guidelines of 10 CFR 50, Appendix K, for evaluating the emergency-core-cooling-system (ECCS) during a postulated LOCA. The framework of the evaluation model for B&W plants is developed in Ref.23 and the primary analytical tools used in this analysis are the CRAFT2[24], REFLOD3[25], FLECSET[26], THETA1-B[27], and TACO3[1] computer codes. The evaluation model for recirculating steam generator PWR plants is presented in Ref.28 and the computer codes used include RELAP5/MOD2-B&W[29], REFLOD3[25], FRAP-T6-B&W[30], BEACH[31], and TACO3[1].

ECCS analyses were conducted to determine the impact on the LOCA limits of a typical B&W Mk-B (15x15) fuel design operating with an internal gas pressure that exceeds the nominal RCS pressure. These analyses were conducted for B&W 177-FA lowered loop plants operating at 2772 MWt. The most limiting break size and location were used in these analyses.

[b,c,d,e]

[b,c,d,e]

The analyses described above represents an example of the method that will be used to determine the LOCA limits when the fuel rod internal gas pressure exceeds the nominal RCS pressure. Similar analyses will be performed, as required, to determine the impact on LOCA limits for other plants and rod designs. If the higher rod pressures have a negative impact on cladding rupture or PCT during a LOCA, the LOCA linear heat rate limit will be reduced accordingly. The fuel performance code input parameters and assumptions will be verified on a plant-cycle specific basis during fuel reload analyses to ensure that the LOCA limits are applicable.

Figure 5.1 LOCA Peak Linear Heat Rates
Typical Mk-B (15x15) Fuel Design

[b,c,d,e]

6.0 APPLICATION METHODOLOGY

The revised fuel rod internal gas pressure criterion requires that "the internal pressure of the peak fuel rod in the reactor will be limited to a value below that which would cause (1) the fuel-clad gap to increase due to cladding creep during steady-state operation and (2) extensive DNB propagation to occur." This revised criterion replaces the current criterion[1,2] which limits the internal gas pressure to less than the nominal RCS pressure. The manner in which the revised criterion will be applied to design and licensing analyses is presented below.

6.1 Fuel-Cladding Gap Stability

Fuel-cladding gap stability is necessary to ensure that cladding integrity and subchannel flow characteristics are maintained within acceptable limits during operation. The purpose of this subsection is to describe the method used for determining when the cladding diametral creep-out rate will exceed the fuel diametral growth rate. This method will utilize the fuel and cladding diametral strain rate models described in Section 2.

The analysis to determine when fuel-clad lift-off initially occurs will be conducted as follows.

[b,c,d,e]

[b,c,d,e]

[b,c,d,e]

6.2 DNB Propagation Evaluation

The analyses described in Section 4.0 of this report apply directly for Mark B and Mark BW core evaluations provided that the power histories shown in Figure 4.1 and the core design parameters of Tables 4.11 and 4.12 also apply. A core reload evaluation will include a verification check to ensure that these parameters are valid. When reanalysis is necessary, the methods described in Section 4.0 will be used.

7.0 REFERENCES

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APPENDIX A

SCCPB.BAS

A computer code for Statistical Calculation of
Core Protection for the Mark B Core

[b,c,d,e]
Pages A-1 through A-8

APPENDIX B

SCCPBW.BAS

A computer code for Statistical Calculation of
Core Protection for the Mark BW Core

[b,c,d,e]

Pages B-1 through B-9

APPENDIX C

Responses to the NRC Request
For Additional Information

[b,c,d,e]

Pages C-1 through C-29

APPENDIX D

[b,c,d,e]

Pages D-1 through D-3