SECTION J

Westinghouse Electric Corporation
Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

January 22, 1999
NSD-NRC-99-5822

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

Attention: T. E. Collins, Chief
Reactor Systems Branch
Division of Systems Safety and Analysis

Subject: Supplemental Response to NRC "Request for Additional Information for Westinghouse Topical Report WCAP-15063-P, 'Westinghouse In-Reactor Creep Model'," dated September 10, 1998.

Dear Mr. Collins:

Enclosed are three copies of the supplemental response to the NRC's "Request for Additional Information for Westinghouse Topical Report WCAP-15063-P, "Westinghouse In-Reactor Creep Model". This supplemental response provides a response to the two questions (i.e., 2 and 8) that were deferred in our previous response to questions. Westinghouse has completed QA documentation on all information provided in this response and the previous responses (NSD-NRC-98-5808).

Also enclosed are:

1. One (1) copy of the Application for Withholding, AW-99-1316 with Proprietary Information Notice and Copyright Notice.

2. One (1) copy of Affidavit, AW-99-1316.

This submittal contains Westinghouse proprietary information of trade secrets, commercial or financial information which we consider privileged or confidential pursuant to 10 CFR 9.17(a)(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-99-1316 and should be addressed to H. A. Sepp, Manager of Regulatory and Licensing Engineering, Westinghouse Electric Company, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

Henry A. Sepp, Manager
Regulatory and Licensing Engineering

copy to:
T. E. Collins, NRR
P. C. Wen, NRR

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

January 22, 1999
AW-99-1316

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

Attention: T. E. Collins, Chief,
Reactor Systems Branch
Division of Systems Safety and Analysis

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Supplemental Response to NRC "Request for Additional Information for Westinghouse Topical Report WCAP-15063-P, 'Westinghouse In-Reactor Creep Model'," dated September 10, 1998.

Reference: Letter from H. A. Sepp to T. E. Collins, NSD-NRC-99-5822, dated January 22, 1999

Dear Mr. Collins:

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

The proprietary material for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.790, Affidavit AW-99-1316 accompanies this application for withholding, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

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

Correspondence with respect to this application for withholding or the accompanying affidavit should reference AW-99-1316 and should be addressed to the undersigned.

Very truly yours,

Henry A. Sepp, Manager
Regulatory and Licensing Engineering

cc: T. Carter / NRC (5E7)

Proprietary Information Notice

Transmitted herewith are proprietary and non-proprietary versions of documents furnished to the NRC. In order to conform to the requirements of 10 CFR 2.790 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.790(b)(1).

Copyright Notice

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

Henry A. Sepp

Henry A. Sepp, Manager
Regulatory and Licensing Engineering

Sworn to and subscribed
before me this 22 day
of January, 1999.

Janet A. Schwab

Notary Public

Notarial Seal
Janet A. Schwab, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires May 22, 2000
Member, Pennsylvania Association of Notaries

(1) I am Manager, Regulatory and Licensing Engineering, in the Nuclear Services Division, of the Westinghouse Electric Company, a division of CBS Corporation ("Westinghouse") and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Energy Systems Business Units.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked "Supplemental Response to NRC 'Request for Additional Information for Westinghouse Topical Report WCAP-15063-P, 'Westinghouse In-Reactor Creep Model',' dated September 10, 1998," January 22, 1999, for submittal to the Commission, being transmitted by Westinghouse Electric Company (W) letter (NSD-NRC-99-5822) and Application for Withholding Proprietary Information from Public Disclosure, Henry A. Sepp, Westinghouse, Manager Regulatory and Licensing Engineering to the attention of T. E. Collins, Chief, Reactor Systems Branch, Division of Systems Safety and Analysis. The proprietary information as submitted by Westinghouse Electric Company is to provide supplemental response to the NRC's "Request for Additional Information" on the revised Westinghouse Creep Model that is used in the Westinghouse Performance Analysis and Design model (PAD). This information is provided for review and approval of this revised creep model.

This information is part of that which will enable Westinghouse to:

(a) Ensure proper fuel performance of fuel operating in reactors.

(b) Assist customers to obtain license changes resulting from fuel performance modeling.

Further this information has substantial commercial value as follows:

(a) Westinghouse can use this fuel performance modeling capability to further enhance their licensing position over their competitors.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended for developing the enclosed improved core thermal performance methodology.

Further the deponent sayeth not.

Supplemental Response to NRC "Request for Additional Information for Westinghouse Topical Report WCAP-15063-P, 'Westinghouse In-Reactor Creep Model'," dated September 10, 1998

Question 2: **Please provide the temperature, flux, fluence and stress range to which this model will be applied.**

Response 2: *The following table lists the parameter ranges for a typical reload analysis:*

Parameter	Range
Clad Average Temperature:	[ ]$^{a, c}$
Effective Stress Range:	[ ]$^{a, c}$ [ ]$^{a, c}$
Fluence:	[ ]$^{a, c}$
Flux	[ ]$^{a, c}$

[

]$^{a, c}$

Question 8 **The data used to determine the coefficients to the Westinghouse creep-model for Imp Zr-4 and ZIRLO™ are based on creep due to compressive stresses while the internal rod pressure analysis at end-of-life calculates creep due to tensile stresses. Please provide cold-worked Zr-2 or Zr-4 tube data that demonstrates that compressive in-reactor creep data can be used to predict in-reactor creep under tensile stresses.**

Response: *The evaluation that resulted in the determination that creep in tension and compression are the same was based on in-reactor creep data measured in tension and compression. Two investigations[(1)(2)] have studied the irradiation creep behavior of Zircaloy subject to compressive and tensile stresses. Both of these studies used Zircaloy tubing and concluded that irradiation creep in tension is greater than compression. Specifically, one investigation concluded for steady-state irradiation creep that tensile creep is 1.7 times greater than compressive creep[(2)].*

However, [

]a, c and will show that irradiation creep is the same in compression and tension.

<u>*Analysis of the Data Reported by Garzarolli, et al.*</u>[1]

Garzarolli, et al.[1]*, reported Zircaloy biaxial tension irradiation strain data in compression and tension. The reported data included post-irradiation strain measurements on irradiated unstressed and stressed tubes. The strain data were approximately linear with increasing time. [*

]a, c. Siemens[3] *has previously used the thinwall stress equation (at the tube mid-wall location) to calculate the hoop stress for thermal creep data analysis. [*

]a, c

[ ]a, c

[

]a, c

[

]a,c

[ ]a,c

Figure 4 presents a plot of the total strain rate versus the thickwall mid-wall hoop stress. [

]a,c

The value of de/dt(ic) was calculated from de/dt(total) based on the regression fit method. [

]a,c

Analysis of the Data Reported by McGrath[2]

McGrath reported Zr-2 biaxial tension irradiation strain data in compression and tension. The reported data were in-reactor diameter strain measurements performed as a function of time on a recrystallized annealed Zr-2 sample pre-irradiated to a fluence of $6x10^{21}$ n/cm^2 $(E>1$ *MeV) at different stress levels.* [

$]^{a,c}$

The value of de/dt(ic) was calculated from de/dt(total) based on the regression fit method. [

$]^{a,c}$

Evaluation of the Data Reported by Stehle, et al.[(3)]

[

$]^{a,c}$ *Stehle*[(3)] *reported out-reactor CWSR Zircaloy thermal creep data which showed that creepout is significantly greater than creepdown.* [

$]^{a,c}$

The [$]^{a,c}$ *relationship between the hoop and equivalent stresses is,*

[$]^{a,c}$

and the [$]^{a,c}$ *hoop stress used by Stehle*[(3)] *is,*

[$]^{a,c}$

[

]$^{a, c}$

[ ]$^{a, c}$

[

]$^{a, c}$

Conclusions

The above analysis and discussion of the results show that the irradiation creep rate is [ ]$^{a, c}$

References

1. *F. Garzarolli, H. Stehle and E. Steinberg, "Behavior and Properties of Zircaloys in Power Reactors: A Short Review of Pertinent Aspects in LWR Fuel," ASTM STP 1295, ASTM, 1996, pp. 12-32.*

2. *M. A. McGrath, Annual Meeting on Nuclear Technology '98, the German Nuclear Society and the German Atomic Forum, Munchen Park Hilton, Munich, Germany, May 26-28, 1998.*

3. *H. Stehle, E. Steinberg and E. Tenckhoff, "Mechanical Properties, Anisotropy, and Microstructure of Zircaloy Canning Tubes," Zirconium in the Nuclear Industry, ASTM STP 633, ASTM, 1977, pp. 486-507.*

4. *T.H. Lin, "Theory of Inelastic Structures," 1968.*

Table 1

Zircaloy Biaxial Irradiation Creep Rates

for Tension and Compression

a, c

Table 2

Thinwall and Thickwall Hoop Stress Values at the Mid-wall

a, c

Figure 1

In-Reactor Zircaloy Growth - Large Scatter

0 MPa (based on Garzarolli, et al., ASTM STP 1295 data)

a, c

Figure 2

In-Reactor Biaxial Tension Zircaloy Creep

-83 MPa (-12 ksi) Thinwall Hoop Stress (based on Garzarolli, et al., ASTM STP 1295 data)

a, c

Figure 3

In-Reactor Biaxial Tension Zircaloy Creep

+44 MPa (+6.4 ksi) Thinwall Hoop Stress (based on Garzarolli, et al., ASTM STP 1295 data)

a, c

Figure 4

Zircaloy In-Reactor Creepdown Versus Creepout

(based on Garzarolli, et al., ASTM STP 1295 data)

a, c

Figure 5

Zircaloy In-Reactor Creepdown Versus Creepout

(based on Garzarolli, et al., ASTM STP 1295 data)

a, c

Figure 6.

Zircaloy Creepdown Versus Creepout

(based on Garzarolli, et al., ASTM STP 1295 data)

a, c

Figure 7

RXA Zr-2 Creepdown Versus Creepout

(based on McGrath, Annual Meeting on Nuclear Technology '98 data)

a, c

Figure 8

RXA Zr-2 Creepdown Versus Creepout

(based on McGrath, Annual Meeting on Nuclear Technology '98 data)

a, c

Figure 9

RXA Zr-2 In-Reactor Creepdown Versus Creepout

(based on McGrath, Annual Meeting on Nuclear Technology '98 data)

a, c

Figure 10

Zircaloy Creep in Tension and Compression

a, c

SECTION K

Westinghouse Electric Corporation **Energy Systems**

Box 355
Pittsburgh Pennsylvania 15230-0355

February 5. 1999
NSD-NRC-99-5825

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

Attention: T. E. Collins. Chief
Reactor Systems Branch
Division of Systems Safety and Analysis

Subject: Notification of "Errata for WCAP-15063-P. 'Westinghouse In-Reactor Creep Model'.".

Dear Mr. Collins:

Enclosed are three copies of the Notification of "Errata for WCAP-15063-P. "Westinghouse In-Reactor Creep Model". This notification is to advise the NRC of corrections that will be made to the final approved version of the WCAP. once approval has been received from the NRC. The errata results from changes that ensued as a result of response to "Requests for Additional Information" (RAIs). Please note that this information is advanced notification only.

Also enclosed are:

1. One (1) copy of the Application for Withholding. AW-99-1322 with Proprietary Information Notice and Copyright Notice.

2. One (1) copy of Affidavit. AW-99-1322.

This submittal contains Westinghouse proprietary information of trade secrets. commercial or financial information which we consider privileged or confidential pursuant to 10 CFR 9.17(a)(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-99-1322 and should be addressed to H. A. Sepp. Manager of Regulatory and Licensing Engineering. Westinghouse Electric Company. P. O. Box 355. Pittsburgh. Pennsylvania 15230-0355.

Very truly yours.

Henry A. Sepp. Manager
Regulatory and Licensing Engineering

copy to:
T. E. Collins. NRR
P. C. Wen. NRR
S. L. Wu. NRR

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

February 5, 1999
AW-99-1322

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

Attention: T. E. Collins, Chief,
Reactor Systems Branch
Division of Systems Safety and Analysis

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: Notification of "Errata for WCAP-15063-P, 'Westinghouse In-Reactor Creep Model'.".

Reference: Letter from H. A. Sepp to T. E. Collins, NSD-NRC-99-5825, dated February 5, 1999

Dear Mr. Collins:

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

The proprietary material for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.790, Affidavit AW-99-1322 accompanies this application for withholding, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

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

Correspondence with respect to this application for withholding or the accompanying affidavit should reference AW-99-1322 and should be addressed to the undersigned.

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Henry A. Sepp, Manager
Regulatory and Licensing Engineering

cc: T. Carter / NRC (5E7)

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Copyright Notice

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

Henry A. Sepp

Henry A. Sepp, Manager
Regulatory and Licensing Engineering

Sworn to and subscribed
before me this 5th day
of February, 1999.

Janet A. Schwab
Notary Public

Notarial Seal
Janet A. Schwab, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires May 22, 2000
Member, Pennsylvania Association of Notaries

(1) I am Manager, Regulatory and Licensing Engineering, in the Nuclear Services Division, of the Westinghouse Electric Company, a division of CBS Corporation ("Westinghouse") and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Energy Systems Business Units.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked "Notification of 'Errata for WCAP-15063-P, 'Westinghouse In-Reactor Creep Model'.'," February 5, 1999, for submittal to the Commission, being transmitted by Westinghouse Electric Company (W) letter (NSD-NRC-99-5825) and Application for Withholding Proprietary Information from Public Disclosure, Henry A. Sepp, Westinghouse, Manager Regulatory and Licensing Engineering to the attention of T. E. Collins, Chief, Reactor Systems Branch, Division of Systems Safety and Analysis. The proprietary information as submitted by Westinghouse Electric Company is to advise the NRC of corrections that will be made to the final approved version of the WCAP, once approval has been received from the NRC. The errata results from changes that ensued as a result of response to "Requests for Additional Information" (RAIs).

This information is part of that which will enable Westinghouse to:

(a) Ensure proper fuel performance of fuel operating in reactors.

(b) Assist customers to obtain license changes resulting from fuel performance modeling.

Further this information has substantial commercial value as follows:

(a) Westinghouse can use this fuel performance modeling capability to further enhance their licensing position over their competitors.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended for developing the enclosed improved core thermal performance methodology.

Further the deponent sayeth not.

Errata for WCAP-15063-P

"Westinghouse In-Reactor Creep Model"

Errata for WCAP-15063-P

"Westinghouse In-Reactor Creep Model"

Based on the responses to the Requests for Additional Information (RAIs) on WCAP-15063-P, "Westinghouse In-Reactor Creep Model", that were received from the NRC on September 10, 1998, the following is a list of revisions to WCAP-15063-P. The text to be inserted and to be deleted are enclosed by quotation marks. These revisions will be made to the approved version of the WCAP, once approval has been received from the NRC.

1. Coefficient C_3:

An editorial change is necessary to be made to WCAP-15063-P in order that the method used to evaluate the coefficient C_3 is clearly presented. The evaluation of C_3 is correctly described in the last paragraph of Page 14 but not correctly described in the middle of Page 6. The editorial change involves the deletion of one sentence and the insertion of several sentences on Page 6. The sentence to be deleted on Page 6 is:

"The equation was [ ]a, c IMP Zr-4 out-of-reactor (laboratory) thermal creep tests [ ]a, c and the out-of-reactor (laboratory) creep rate equation for [ ]a, c according to:

[ ]a, c (9)"

The sentences to be inserted in place of the deleted sentence are:

"The equation was [ ]a, c out-of-reactor (laboratory) thermal creep rates [ ]a, c. The calculation was performed for typical fuel rod parameters. The parameters used were [ ]a, c. C_3 was evaluated according to:

[ ]a, c (9)

The results of the analysis are listed in Table 7."

Table 7

Calculated values of the coefficient C_3

T (K)	C_3
589	[ ]a, c
616	[ ]a, c
644	[ ]a, c
672	[ ]a, c
avg. =	[ ]a, c

2. Irradiation Hardening:

An evaluation of the diameter strain exhibited by a cold-worked (CW) Zr-2 pressure tube shows that the Zr-2.5Nb data are applicable to Zr-4. When WCAP-15063-P was prepared, the CW Zr-2 pressure tube data were not known. The availability of the CW Zr-2 pressure tube strain data strengthens the conclusion that the Zr-2.5Nb data are applicable to Zr-4. Section 3.2.1.2 in WCAP-15063-P presents [

] a, c. The CW Zr-2 pressure tube may be added to WCAP-15063-P by modifying one sentence on Page 11 in the first paragraph of Section 3.2.1.2, adding text just after the modified sentence, modifying Table 3, adding one Figure A, and adding one reference. The sentence to be modified is:

"[

] a, c "

The modified sentence is:

"[

] a, c "

The text to be added is:

"[

] a, c. This shows that the Zircaloy-2.5Nb data are applicable to Zircaloy-4."

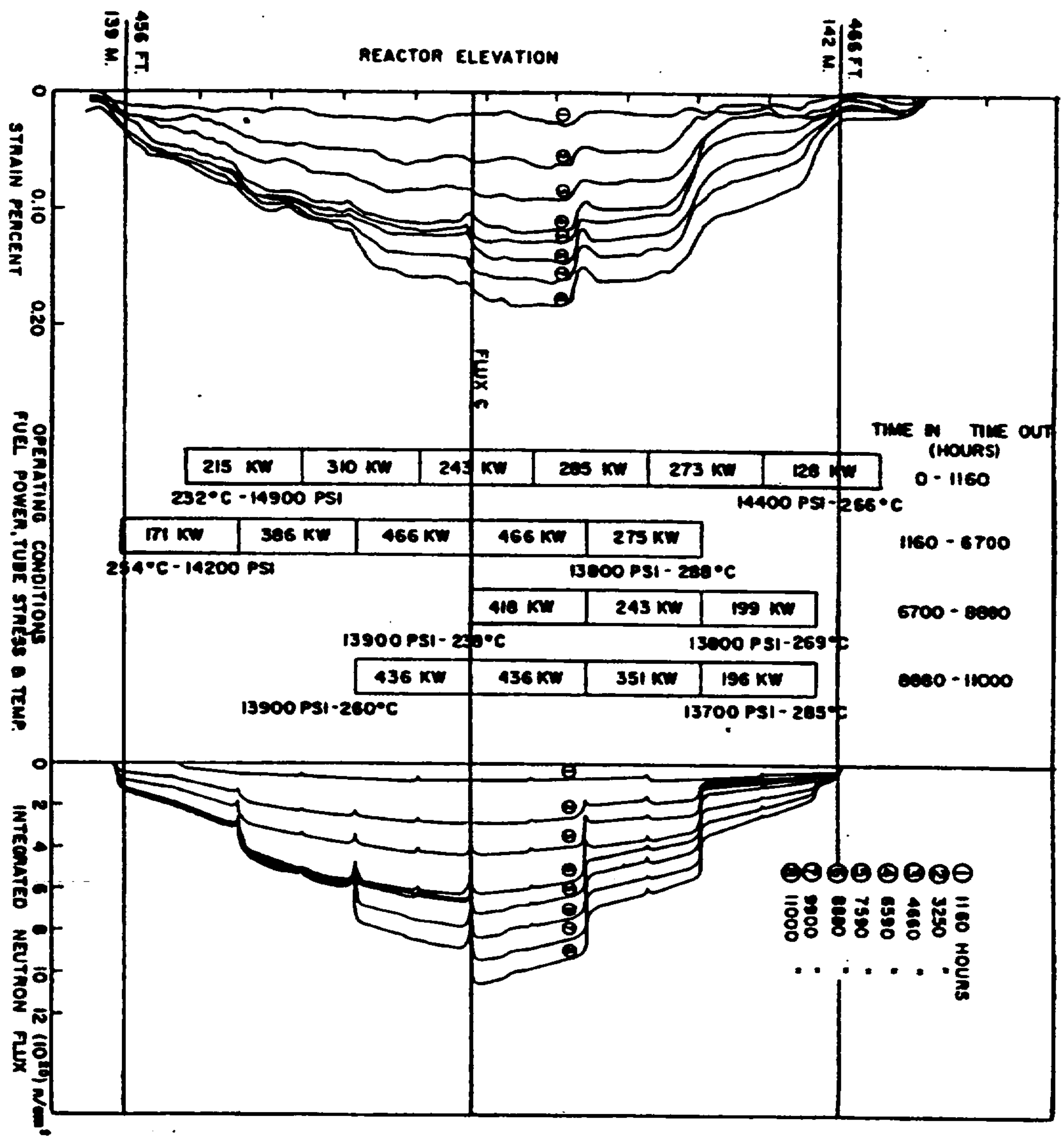
Reference:

18. P. A. Ross-Ross and C. E. L. Hunt, "The In-Reactor Creep of Cold-Worked Zircaloy-2 and Zirconium-2.5 wt % Niobium Pressure Tubes," Journal of Nuclear Material, Volume 26, 1968, pages 2 through 17.

Table 3
Irradiation Hardening Reduction Factors Evaluated by
Strain Ratio and Strain Rate Ratio Analysis

a, c

Figure 18



Reprinted from P. A. Ross-Ross and C. E. L. Hunt, "The In-Reactor Creep of Cold-Worked Zircaloy-2 and Zirconium-2.5 wt % Niobium Pressure Tubes," Journal of Nuclear Materials, Volume 26, 1968, pages 2 through 17, with permission from Elsevier Science

3. Typographical Errors:

- The date of reference 12 is incorrect. Change the date for Reference 12 from "1974" to "1993".

- The saturated primary strain in Equation 14 does not include the superscript. For the left hand side of Equation 14, change " e_p " to " e_p^{sat} ".

4. Thermal Creep Equations:

The in-reactor thermal creep equations presented in WCAP-15063-P represent asymptotic approximations. These approximations were used for simplicity. After preparation of WCAP-15063-P, the equations were coded into PAD, and the exact equations were used. Therefore, the exact equations used by PAD need to be included in WCAP-15063-P. This may be accomplished by completely replacing section 2.1 with the following.

"2.1 In-Rector Thermal Creep Overview

The new in-reactor creep model was developed to describe Westinghouse cold-worked stress relieved (CWSR) improved (IMP) Zr-4 (low tin Zr-4) tubing. The model is based on [

]^{a,c}. The in-reactor thermal creep is given by the out-of-reactor (laboratory) thermal creep corrected for in-reactor irradiation hardening. This behavior is described by:

[ ]^{a,c} (3)

where [

]^{a,c}. The equation
[
]^{a,c}.
The equation for [ ]^{a,c} is given by:

[ ]^{a,c} (11)

where [
]^{a,c}. The thermal creep strain given by [
]^{a,c} IMP Zr-4 thermal creep tests [ ]^{a,c}
according to:

[ ]^{a,c} (4)

The expression for IH is:

[ ]^{a,c} (5)

where [ ]^{a,c}. Equation (5) provides a smooth transition with increasing fluence from no irradiation hardening [ ]^{a,c} to complete irradiation hardening [ ]^{a,c}. The irradiation hardening [ ]^{a,c}. The application of the [ ]^{a,c}.

Since both [ ]^{a,c}. Thus:

[ ]^{a,c} (6)

The derivative of the numerator is given by:

[ ]^{a,c} (35)

The derivative of the denominator is given by:

[ ]^{a,c} (36)

and the expression for de/dt(tc) is:

[ ]^{a,c} (37)"

5. Correlation of In-Reactor with Out-Reactor Creep:

The out-of-reactor (laboratory) thermal creep rates are directly related to the in-reactor irradiation creep rates for a given zirconium alloy. This relationship [ ]^{a,c}. WCAP-15063-P presents the analysis of the [ ]^{a,c}. The Westinghouse BR-3 ZIRLO™ fuel rod data and Westinghouse North Anna ZIRLO™ fuel rod data may be included by deleting one sentence and adding two paragraphs. The sentence to be deleted is the last sentence of the 2nd paragraph on Page 14:

"[

]^{a,c}."

The following two paragraphs are to be added at the beginning of Section 3.2.2.2, Normalization of B&W/EPRI Irradiation Creep to Westinghouse Behavior:

"The out-of-reactor (laboratory) thermal creep rates are directly related to the in-reactor irradiation creep rates for a given zirconium alloy. This relationship [

]^{a,c}.

In the case of the [

]^{a,c}."

Table 8

ZIRLO™ North Anna In-Reactor Creepdown

and Out-of-Reactor Thermal Creepout

Rod	Burnup (MWD/MTU)	Average Creepdown (mils)	Lot	Sample	Steady State Creep Rate (10^{-3} %/h)

a, c

Figure 19

Westinghouse BR-3 Fuel Rod Profilometry (Creepdown)

ZIRLO™ 15x15 Cladding

a, c

Figure 20

Out-of-Reactor Thermal Creep (Creepout)

658 K (725 °F), 108 MPa (15.6 ksi) Equivalent Stress

a, c

Figure 21

Comparison of In-Reactor and Out-of-Reactor Creep Rates

Westinghouse ZIRLO™

a, c

6. Creep Model Predictions:

The text describing the predictions of the in-reactor creep model were inadvertently omitted. The following section should be added:

"3.3 In-Reactor Creep Model

The behavior of the in-reactor creep model is illustrated in Figure 17. The calculation of the creep rates were performed for typical fuel rod parameters. The parameters were [

]^{a,c}. The out-of-reactor thermal and in-reactor thermal creep components are also shown."

SECTION L

Westinghouse
Electric Company

Box 355
Pittsburgh Pennsylvania 15230-0355

November 18, 1999
NSBU-NRC-99-5956

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

Attention: J. S. Wermiel, Chief
Reactor Systems Branch
Division of Systems Safety and Analysis

Subject: Westinghouse Improved Performance Analysis and Design Model (PAD 4.0), WCAP-15063-P, Revision 1 (Proprietary)

Dear Mr. Wermiel:

Enclosed are copies of the Proprietary and Non-Proprietary versions of the following documents. These documents comprise the final submittal package for the "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)".

Attachment 1. Re-write of WCAP-15063-P, as requested by Mr. R. Caruso of the NRC during the June 23, 1999 meeting. The re-write comprises the original topical report submittal as Section 1, "Westinghouse In-Reactor Creep Model" and the "Other PAD Model Changes", that were presented to the NRC during the September 15, 1998 meeting, as Section 2. The "Other PAD Model Changes" were documented in NSD-NRC-98-5810, November 13, 1998. As directed by Mr. R. Caruso, the topical report number remains the same (e.g., WCAP-15063-P); however, the title has been revised to more accurately account for all changes.

Attachment 2. WCAP-15063-P Errata for Section 1 of the re-write topical report. These corrections have been noted to the NRC in response to other RAIs. The corrections to pages are specifically identified and will be incorporated in the base document when the approved version of the WCAP is prepared.

Attachment 3. Response to Question #9 of the NRC's Request for Additional Information on PAD Model Revisions based on the NRC/Westinghouse meeting held on September 15, 1998. This response to RAI #9 is the last outstanding RAI on the PAD topical report submittal.

Attachment 4. Final PAD 4.0 Calibration and V&V Data Package. This calibration and V&V package supersedes the Calibration and V&V Data Package supplied in NSD-NRC-98-5787, September 11, 1998.

Attachment 5 Discussion of PAD 4.0 implementation and requested SER language regarding "Legacy Fuel" and PAD 4.0 implementation.

Also enclosed are:

1. One (1) copy of the Application for Withholding, AW-1371 with Proprietary Information Notice and Copyright Notice.

2. One (1) copy of Affidavit, AW-1371.

It is also requested that a follow-up meeting be held between the NRC and Westinghouse regarding this final submittal package to address any minor questions that may exist, to summarize each of the attachments provided in this submittal, and to discuss schedules. It is planned that this meeting will be held during the week of December 6th, 1999.

This submittal contains Westinghouse proprietary information of trade secrets, commercial or financial information which we consider privileged or confidential pursuant to 10 CFR 9.17(a)(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-1371 and should be addressed to H. A. Sepp, Manager of Regulatory and Licensing Engineering, Westinghouse Electric Company, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

H. A. Sepp

Henry A. Sepp, Manager
Regulatory and Licensing Engineering

Copy to:
S. L. Wu, NRR
R. Caruso, NRR
S. Bloom, NRR

Westinghouse
Electric Company

Box 355
Pittsburgh Pennsylvania 15230-0355

November 18, 1999
AW-1371

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

Attention: J. S. Wermiel, Chief,
Reactor Systems Branch
Division of Systems Safety and Analysis

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: Westinghouse Improved Performance Analysis and Design Model (PAD 4.0), WCAP-15063-P, Revision 1 (Proprietary)

Reference: Letter from H. A. Sepp to J. S. Wermiel, NSBU-NRC-99-5956, dated November 18, 1999

Dear Mr. Wermiel:

The application for withholding is submitted by Westinghouse Electric Company LLC, a Delaware limited liability company ("Westinghouse"), pursuant to the provisions of paragraph (b)(1) of Section 2.790 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary material for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.790, Affidavit AW-1371 accompanies this application for withholding, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

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

Correspondence with respect to this application for withholding or the accompanying affidavit should reference AW-1371 and should be addressed to the undersigned.

Very truly yours,

Henry A. Sepp, Manager
Regulatory and Licensing Engineering

cc: T. Carter / NRC (5E7)

Proprietary Information Notice

Transmitted herewith are proprietary and non-proprietary versions of documents furnished to the NRC. In order to conform to the requirements of 10 CFR 2.790 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.790(b)(1).

Copyright Notice

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

Henry A. Sepp, Manager
Regulatory and Licensing Engineering

Sworn to and subscribed
before me this 18th day
of November, 1999.

Notary Public

Notarial Seal
Janet A. Schwab, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires May 22, 2000
Member, Pennsylvania Association of Notaries

JANET A. SCHWAB
COMMONWEALTH OF PENNSYLVANIA
NOTARY PUBLIC

(1) I am Manager, Regulatory and Licensing Engineering, in the Nuclear Services Division, of the Westinghouse Electric Company LLC, a Delaware limited liability company ("Westinghouse") and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Electric Company.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0), WCAP-15063-P" (Proprietary), November 18, 1999, for submittal to the Commission, being transmitted by Westinghouse Electric Company (W) letter (NSBU-NRC-99-5956) and Application for Withholding Proprietary Information from Public Disclosure, Henry A. Sepp, Westinghouse, Manager Regulatory and Licensing Engineering to the attention of J. S. Wermiel, Chief, Reactor Systems Branch, Division of Systems Safety and Analysis. The proprietary information is the final submitted package by Westinghouse Electric Company to the NRC staff in support of topical report , WCAP-15063-P.

This information is part of that which will enable Westinghouse to:

(a) Ensure proper fuel performance of fuel operating in reactors.

(b) Assist customers to obtain license changes resulting from fuel performance modeling.

Further this information has substantial commercial value as follows:

(a) Westinghouse can use this fuel performance modeling capability to further enhance their licensing position over their competitors.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended for developing the enclosed improved core thermal performance methodology.

Further the deponent sayeth not.

Attachment 1

Proprietary and Non-Proprietary Versions

Of

Re-write of WCAP-15063-P

"Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)"

Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)

June 1998 original
November 1999 revision

Authors:
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Westinghouse Improved Performance Analysis and Design Model
(PAD 4.0)
Executive Overview

This revised topical report addresses all model changes made to the Westinghouse Performance Analysis and Design (PAD) model. The original topical report, submitted to the NRC in June 1998, addressed only changes made to the creep model used in the PAD model, but as the model development progressed, additional changes were identified. These additional changes were formally presented to the NRC in September 1998. Since all of these changes have been formally reviewed by the NRC, it has been requested by the NRC that the originally submitted topical report be revised to encompass all the model changes made to PAD. Therefore, the revised topical report has been divided into two sections: Section 1 (the originally submitted in-reactor creep model), and Section 2 (other model changes made to PAD and submitted to the NRC in September 1998).

PAD is a best estimate fuel rod performance model used for both fuel rod performance and safety analysis inputs. The last version of PAD that was reviewed by the NRC was PAD 3.4. The changes made to PAD 4.0 will be related to changes made from this previously licensed version (i.e., PAD 3.4).

Section 1

Westinghouse In-Reactor Creep Model

Westinghouse In-Reactor Creep Model

1.0 Introduction & Background

The Westinghouse Performance Analysis and Design (PAD) model is a best estimate fuel rod performance model used for both fuel rod performance analysis and safety analysis inputs. The PAD code consists of several fuel rod performance models integrated to predict fuel temperature, rod pressure, fission gas release, cladding elastic and plastic behavior, cladding growth, cladding corrosion, fuel densification, and fuel swelling as a function of linear power and time. Many of the fuel rod performance models were first introduced to the NRC (then AEC) in the 1972-1973 time frame[1][2][3][4]. Subsequent to the original model introduction, two specific revisions have been submitted for review and approval (i.e., PAD 3.3[5] and PAD 3.4[6]). With respect to the creep model used in PAD, the original model form remains in effect except for a revision that occurred to the irradiation enhanced creep portion of the model in PAD 3.4. The thermal creep portion of the model has remained the same since the model's inception in 1972.

1.1 Purpose

The purpose of this section is to introduce the new creep model to be used in the overall PAD fuel rod performance model. The new creep model accounts for advances in the understanding of in-reactor creep that have occurred between 1972 and 1998, and represents a description of in-reactor creep relative to the information and data that are available in 1998. This model enhancement is projected to restore rod internal pressure limit margin to the fuel rod design criterion.

1.2 Discussion of the Current PAD Creep Model (PAD 3.4)

The total in-reactor creep rate, de/dt, in PAD 3.4 is evaluated as the sum of the out-of-reactor (laboratory) thermal creep rate, de/dt(out-rx tc), plus the irradiation enhanced creep rate, de/dt(ic).

$$de/dt = de/dt(out-rx\ tc) + de/dt(ic) \qquad (1)$$

The out-of-reactor (laboratory) thermal creep rate, de/dt(out-rx tc), is a function of clad temperature, clad equivalent or effective stress and time. [

]^{a,c}.

The irradiation enhanced creep rate, de/dt(ic), is a function of neutron flux and fluence. [

]$^{a, c}$.

1.3 Evaluation of PAD 3.4 Creep Model - Need for Change

With the current generation of fuel and the enhanced operational performance requirements placed on the fuel (i.e., increased cycle lengths, higher operating system temperatures, higher operating power levels, higher peaking factors, and higher burnup levels), enhanced modeling and prediction capabilities are necessary to demonstrate the continued acceptable performance of the fuel to the original fuel rod design criteria. As such, new post-irradiation examination (PIE) data needs to be accounted for and incorporated into the fuel rod performance models. This new PIE data has already demonstrated a need to revise the fuel rod corrosion model in PAD(7). In addition, other material property characteristics exist that previously had not been accounted for, either due to the lack of available data or the level of sophistication of the mechanics. With new data now available and the level of sophistication of the mechanics reaching closer to the phenomenological level, significant improvements to the fuel rod performance models can be achieved.

A review of current in-reactor creep models and methods was performed by Westinghouse relative to the state-of-the art mechanics of fuel rod behavior. This involved a detailed review of work performed by AECL and reported in 1996 by Christodoulou et al.(8). The subsequent work performed at AECL, reported by Christodoulou, has demonstrated that the PAD 3.4 in-reactor creep model is overly conservative and needs to be revised. Christodoulou presented the formulation and results of a fundamental-empirical model describing the in-reactor creep of cold-worked (CW) Zr-2.5Nb for pressure tube application. Some of the many models proposed to describe the in-reactor creep of zirconium alloys are described in References 8, 9 and 10. The Christodoulou model is considered to be the most fundamental model that is also based on the largest in-reactor data set to date. The model includes the effects of texture, grain shape, anisotropy and the relative contributions of prismatic, basal and pyramidal planes to dislocation climb assisted glide. This in-reactor model includes data from creep measurements of pressure tubes in power reactors, pressure tubes in test reactors, small pressurized tubes in test reactors and beam stress relaxation samples in test reactors. The test data includes samples with thermal creep strain. In addition, the test data includes textures typical of both pressure and fuel-cladding tubes. As

a result, the framework of this model was selected by Westinghouse to formulate a new in-reactor creep model for fuel rod application.

According to the Christodoulou model, the in-reactor creep rate is the sum of the in-reactor thermal creep rate, de/dt(tc), and the irradiation enhanced creep rate, de/dt(ic).

$$de/dt = de/dt(tc) + de/dt(ic) \qquad (2)$$

[

]^{a,c}. As a result, the predicted total creep rate from the current PAD model (Equation (1)) is higher than that derived from Equation (2) and is therefore conservative. This effect will be discussed in subsequent sections.

The PAD creep model needs to be revised due to the demonstrated fact that the original PAD 3.4 creep model is conservative; therefore, there is a need to account for new PIE data and material property behavior. Specifically, the PAD 3.4 in-reactor creep model is being replaced for the following reasons:

- [

  ]^{a,c}
- [ ]^{a,c}
- [

  ]^{a,c} Out-of-reactor and in-reactor creep behavior is dependent on fabrication process parameters such as final area reduction, intermediate anneal temperature, final anneal temperature and time and post-extrusion anneal temperature.
- [ ]^{a,c}
- [

  ]^{a,c}

2.0 New PAD In-Reactor Creep Model

The new in-reactor creep model developed for fuel rod application in PAD is based on [

]$^{a, c}$.

According to Christodoulou, the total in reactor creep rate is the sum of the in-reactor thermal creep rate, de/dt(tc), and the irradiation enhanced creep rate, de/dt(ic)

$$de/dt = de/dt(tc) + de/dt(ic) \tag{2}$$

[ ]$^{a, c}$.

The new in-reactor creep model was developed to describe Westinghouse cold-worked stress relieved (CWSR) tubing. The specific material behavior of the new PAD model is based on Westinghouse cladding. [

]$^{a, c}$ the new creep model describe Westinghouse cladding.

2.1 In-Reactor Thermal Creep Overview

The in-reactor thermal creep component was developed using Westinghouse cold-worked stress relieved (CWSR) improved (IMP) Zr-4 (low tin Zr-4) tubing. The in-reactor thermal creep is given by the out-of-reactor (laboratory) thermal creep corrected for in-reactor irradiation hardening. This behavior is described by:

[ ]$^{a, c}$ (3)

where [

]$^{a, c}$. The equation [

]$^{a, c}$. The equation for [ ]$^{a, c}$ is given by:

[ ]$^{a, c}$ (4)

where [ ]$^{a, c}$. The equation was [ ]$^{a, c}$ IMP Zr-4 thermal creep tests [ ]$^{a, c}$ according to:

[ ]$^{a, c}$ (5)

The irradiation hardening [ ]$^{a, c}$. The expression for IH is:

[ ]$^{a, c}$ (6)

where [ ]$^{a, c}$. Equation (6) provides a smooth transition with increasing fluence from no irradiation hardening [ ]$^{a, c}$ to complete irradiation hardening [ ]$^{a, c}$. The application of the [ ]$^{a, c}$ was supported by an evaluation of the creep activation energy (discussed below) and the in-reactor thermal creep hardening model reported by Limback and Andersson[10] for CWSR Zr-4.

2.2 Irradiation Enhanced Creep Overview

The new irradiation enhanced creep component was developed using [ ]$^{a, c}$. The irradiation creep behavior [ ]$^{a, c}$. Since the [ ]$^{a, c}$. The irradiation enhanced creep rate equation is given by:

[ ]$^{a, c}$ (7)

where [

]^{a,c}. The [ ]^{a,c} creep rate equation is given by:

[ ]^{a,c} (8)

where [

]^{a,c}. The [

]^{a,c}. The equation was [ ]^{a,c} IMP Zr-4 out-of-reactor (laboratory) thermal creep tests [

]^{a,c} and the out-of-reactor (laboratory) creep rate equation for [ ]^{a,c} according to:

[ ]^{a,c} (9)

The [ ]^{a,c} measurements were performed at PWR reactor coolant temperature. Irradiation enhanced creep increases with increasing temperature. The [

]^{a,c} in:

[ ]^{a,c} (10)

where [

]^{a,c}. Hence, Equation (10) gives the [

]^{a,c}.

A more detailed evaluation of each component of the PAD model is provided in the subsequent sections.

3.0 Creep Model Detailed Justification

As stated in the previous section, a more detailed justification for the equations and coefficients follows below for a more thorough understanding of Westinghouse's model development.

3.1 Out-of-Reactor (Laboratory) Thermal Creep

The out-of-reactor (laboratory) creep behavior of CWSR Zr-4 tubing fabricated by Westinghouse was established for [ ]$^{a, c}$. Internal pressure creep tests were conducted using [ ]$^{a, c}$. The test samples in each test condition were strained into the secondary creep region. The internal pressure and diametral strain were converted to mid-wall hoop stress and strain. The mid-wall hoop strain data were analyzed by separating the total strain into primary and secondary components. The following equations resulted:

- Total creep strain, e (fraction):

[ ]$^{a, c}$ (11)

where t is the time (hour).

- Secondary creep rate, $(de/dt)_S$ (fraction/hour):

[ ]$^{a, c}$ (12)

where [ ]$^{a, c}$.

- Elastic modulus, E_E (psi):

[ ]$^{a, c}$ (13)

where TF is the temperature in (°F).

- Saturated primary strain, e_p (fraction):

[ ]a, c (14)

- Time coefficient, K:

[ ]a, c (15)

The PAD code calculates [

]a, c

is:

[ ]a, c (16)

The coefficient [

] a, c is therefore given by:

[ ]a, c (17)

3.2 In-Reactor Thermal Creep

3.2.1 Irradiation Hardening

3.2.1.1 Model Development

The determination of the in-reactor creep components may be illustrated by the CW Zr-2.5Nb pressure tube reported by Fidleris[11] as shown in Figure 2. The tube was irradiated for 27,550 hours in the Whiteshell WR-1 test reactor. At the outlet end of the tube the temperature is 650K (711 °F) and the hoop stress is 43 MPa (6.2 ksi), [ ] a, c. These temperatures are considerably higher than normal CANDU pressure tube service operation temperatures, because the Whiteshell test reactor used organic coolant.

[

] a, c. This clearly shows that irradiation reduces the out-of-reactor (laboratory) thermal creep strain, i.e., that irradiation hardening of out-of-reactor (laboratory) thermal creep occurs.

The irradiation hardening of out-of-reactor (laboratory) thermal creep is further illustrated by [

]^{a,c}. This clearly shows that irradiation decreases (or "hardens") the out-of-reactor (laboratory) thermal creep. [

]^{a,c}.

The irradiation hardening effect on the out-of-reactor (laboratory) thermal creep is even noticeable [

]^{a,c}.

The irradiation enhanced component is [

]^{a,c}.

$$[\qquad\qquad\qquad]^{a,c} \qquad (18)$$

This is shown in Figure 3 as the irradiation enhanced component.

The irradiation hardening [

] a, c may be described by an equation of the form:

[ ]a, c (19)

where Φ is the fluence in n/cm^2 ($E > 1$ MeV). Equation (19) provides a smooth transition with increasing fluence from no irradiation hardening [ ] a, c to complete irradiation hardening [ ] a, c.

3.2.1.2 Model Evaluation

The irradiation hardening factor, [ ] a, c. The results were:

[ ]a, c (20)

[ ]a, c

The [ ] a, c. The result was:

[ ]a, c (21)

These results indicate that the in-reactor irradiation hardening of thermal creep is [ ] a, c.

Limback and Andersson[10] reported a model that describes the in-reactor creep behavior of CWSR Zr-4 cladding. The [

] a, c. The

equations are:

[ ]a, c (22)

where [ ] a, c, and:

[ ])]a, c (23)

where [

] a, c Equation (23) becomes:

[ ]a, c (23a)

The calculated IH factors using Equation (23a) are presented in Figure 6 as a dashed line. Figure 6 shows that the calculated [

] a, c.

3.2.2 Irradiation Creep

3.2.2.1 Modeling of the B&W/EPRI Data

The determination of the irradiation enhanced creep component was performed using the reported B&W/EPRI Oconee-2 creepdown data[15]. The tabulation presented by Franklin[20] is the [

] a, c. The hoop strain, $\Delta D/D$, was described by an equation of the form:

[ ]a, c (24)

where [

] a, c:

[ ] a, c (25)

[ ] a, c

Figure 12 shows that this fit is in excellent agreement with the data.

[

] a, c.

The steady state irradiation creep component is [

] a, c.

The [

] $^{a, c}$.

3.2.2.2 Normalization of B&W/EPRI Irradiation Creep to Westinghouse Behavior

The B&W/EPRI in-reactor creep data [

] $^{a, c}$.

The out-of-reactor (laboratory) thermal creep rates may be [

] $^{a, c}$.

The irradiation enhanced creep behavior [

] $^{a, c}$ according to Equation (26):

[ ]a, c (26)

[

]a, c:

[ ]a, c (27)

Equation (27) may be written as:

[ ]a, c (28)

For [

]a, c. Equation (28) becomes,

[ ]a, c (29)

which is the form of Equation (26). Hence, Equations (26) and (27) are related by the relationships,

[

]a, c. The conversion factors are:

[ ]a, c (30)

[ ]a, c

and the resulting equation for [ ]a, c is:

[ ]a, c (31)

where [

]a, c:

[ ]$^{a, c}$ (32)

The average value for C_3 was

[ ]$^{a, c}$ (33)

This factor was [

]$^{a, c}$.

3.2.2.3 Irradiation Creep Temperature Dependence

The irradiation creep temperature dependence was [

]$^{a, c}$.

The data may be described by a function:

[ ]$^{a, c}$ (34)

where [ ]$^{a, c}$.

4.0 Application to ZIRLO™

The in-reactor creep model developed above to describe CWSR IMP Zr-4 may be applied to ZIRLO™. This application may be accomplished using the Westinghouse IMP Zr-4 and ZIRLO™ fuel rod creepdown data. Generally, after 1-cycle, the cladding is freestanding (i.e., fuel pellet contact has not occurred). [

] a, c.

The irradiation creep behavior exhibited by Westinghouse IMP Zr-4 and ZIRLO™ fuel rods is considered to be consistent with in-reactor irradiation creep data. [

] a, c.

Higher burnup Westinghouse IMP Zr-4 and ZIRLO™ fuel rods are available. Table 6B [

] a, c. As a result of this ZIRLO™ creep discussion, a multiplier will be used to account for ZIRLO™ creep as compared to IMP Zr-4 creep.

5.0 Summary and Conclusions

In summary, the discussion above presented a new in-reactor creep model. The model was developed based on the best available zirconium alloy in-reactor creep models and data available to date. The model is consistent with fundamental descriptions of in-reactor creep. As a result of the mechanistic approach, the model is expected to be much more consistent with in-reactor creep behavior. The model describes the behavior of Westinghouse CWSR tubing. The total in-reactor creep rate is composed of irradiation enhanced and in-rector thermal components. The irradiation enhanced component is dependent on the stress, flux (and fluence) and temperature. The in-reactor thermal component is dependent on the stress, time, temperature and fluence.

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Table 1

IMP Zr-4 Out-of-Reactor (Laboratory) Thermal Creep Data

a, c

Table 2A

Calculated STD Zr-4 Out-of-Reactor Creep Rate Values as a Function of Time

a, c

Table 2B

Out-of-Reactor Creep Rate Normalization Factor for IMP Zr-4 Data Relative to STD

a, c

Table 3

Evaluation of Zr-2.5Nb Out-of-Reactor Thermal Creep

Irradiation Hardening

a

* Minimum diameter near the edge of the fuel.

** Diameter measured about 1 meter from the core edge.

Table 4

Comparison of Oconee-2 In-Reactor and

Out-of-Reactor (Laboratory) Creep Rates.

a, c

Table 5

Average Diameter Creepdown of IMP Zr-4 and ZIRLO™ Fuel Rods

in the High Power Region of North Anna Advanced Material Demonstration Assemblies

a, c

Table 6A

Westinghouse 1-Cycle Fuel Rods

a, c

Table 6B

Westinghouse High Burnup Fuel Rods

a, c

Figure 1

Out-of-Reactor Thermal Creep Steady State Rate - Tin Dependence

a,

Figure 2

CW Zr-2.5Nb Pressure Tube Diameter Data (the flux data are not accurate)

Figure 8 of Reference 11

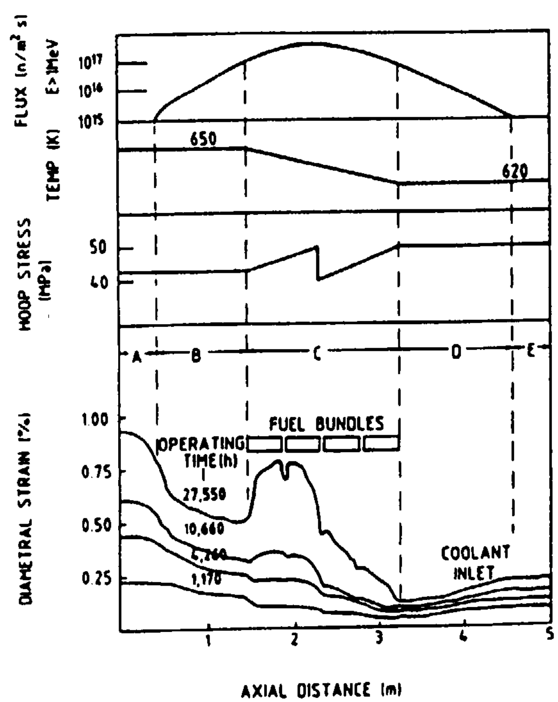


Fig. 8. Diametral creep of cold-worked Zr–2.5Nb loop tube in WR-1 reactor.

Reprinted from Journal of Nuclear Materials, Volume 159,

V. Fidleris, "The Irradiation Creep and Growth Phenomena", pp 22-42,

Figure 3

Creep Components - CW Zr-2.5Nb Pressure Tube

650 K, 43 MPa Hoop Stress

a,

Figure 4

CW Zr-2.5Nb Pressure Tube Diameter Data

Figure 4 of Reference 13

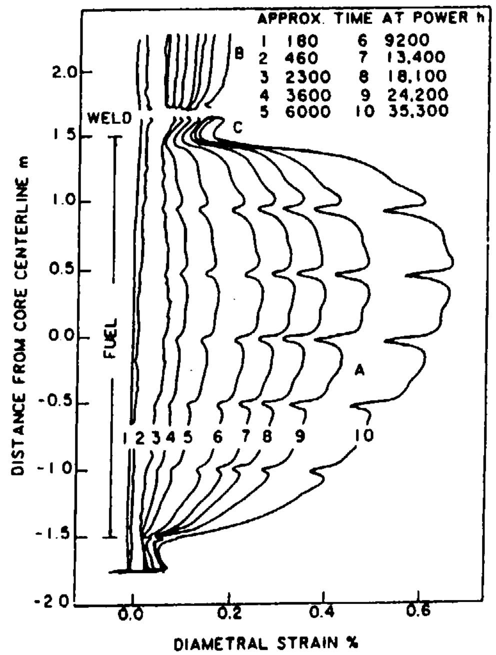


FIGURE 4 Diametral strain of cold-worked Zr-2.5 wt% Nb pressure tube vs axial location in NRU. Illustrating

A approximate cosine distribution of strain in fueled zone
B creep in upper extension, and
C suppression of creep near edge of core.

Reprinted from Dimensional Stability and Mechanical Behaviour Irradiated Metals and Alloys, British Nuclear Energy Society, R. A. Holt, A. R. Causey and V. Fidleris, "Correlation of Creep and Growth of Pressure Tubes with Operating Variables and Microstructure", pp. 175-178, Copyright 1983, with permission from Thomas Telford, London.

Figure 5

CW Zr-2.5Nb Pressure Tube Diameter Data

Figure 7 of Reference 14

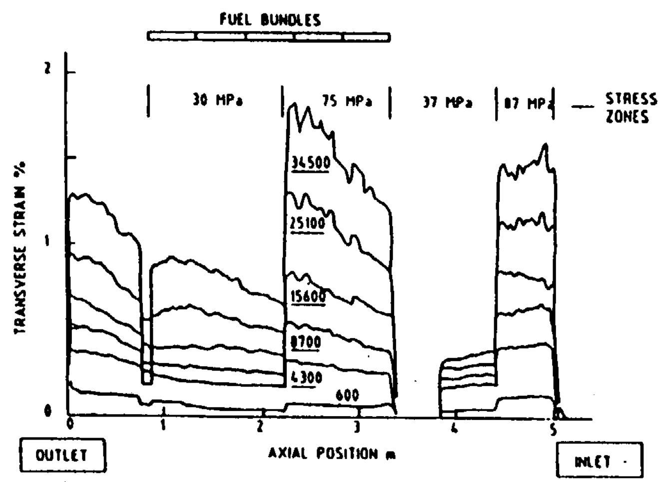


FIG. 7—*Transverse strain profiles of a pressure tube in WR1.*

Reprinted from Influence of Radiation on Material Properties: 13th International Symposium (Part II), ASTM STP 956, A. R. Causey, V. Fidleris, S. R. MacEwen and C. W. Schulte, "In-Reactor Deformation of Zr-2.5 wt% Nb Pressure Tubes", pp 54-68,

Figure 6

Saturated Out-of-Reactor (Laboratory) Thermal Creep Reduction Factor

a,

Figure 7

CWSR Zr-4, B&W/EPRI, Lot S-1, 577-578 K, 69 MPa

a, (

Figure 8

CWSR Zr-4, B&W/EPRI, Lot S-1, 577-578 K, 86 MPa a

Figure 9

CWSR Zr-4, B&W/EPRI, Lot S-1, 577-578 K, 103 MPa

a,

Figure 10

CWSR Zr-4 Saturated Transient Component, B&W/EPRI, Lot S-1, 577-578 K

a, (

Figure 11

CWSR Zr-4 Steady State Component, B&W/EPRI, Lot S-1, 577-578 K

a,

Figure 12

Comparison of Measured and Calculated Strain for CWSR Zr-4

Lot S-1, B&W/EPRI, 577-578 K

a,

Figure 13

CWSR Zr-4, Lot S-1, B&W/EPRI, 581-582 K, 103 MPa a,

Figure 14

CWSR Zr-4 Apparent Temperature Dependence

Lot S-1, B&W/EPRI, 103 MPa

a,

Figure 15

Comparison of In-Reactor and Out-of-Reactor Creep Rates

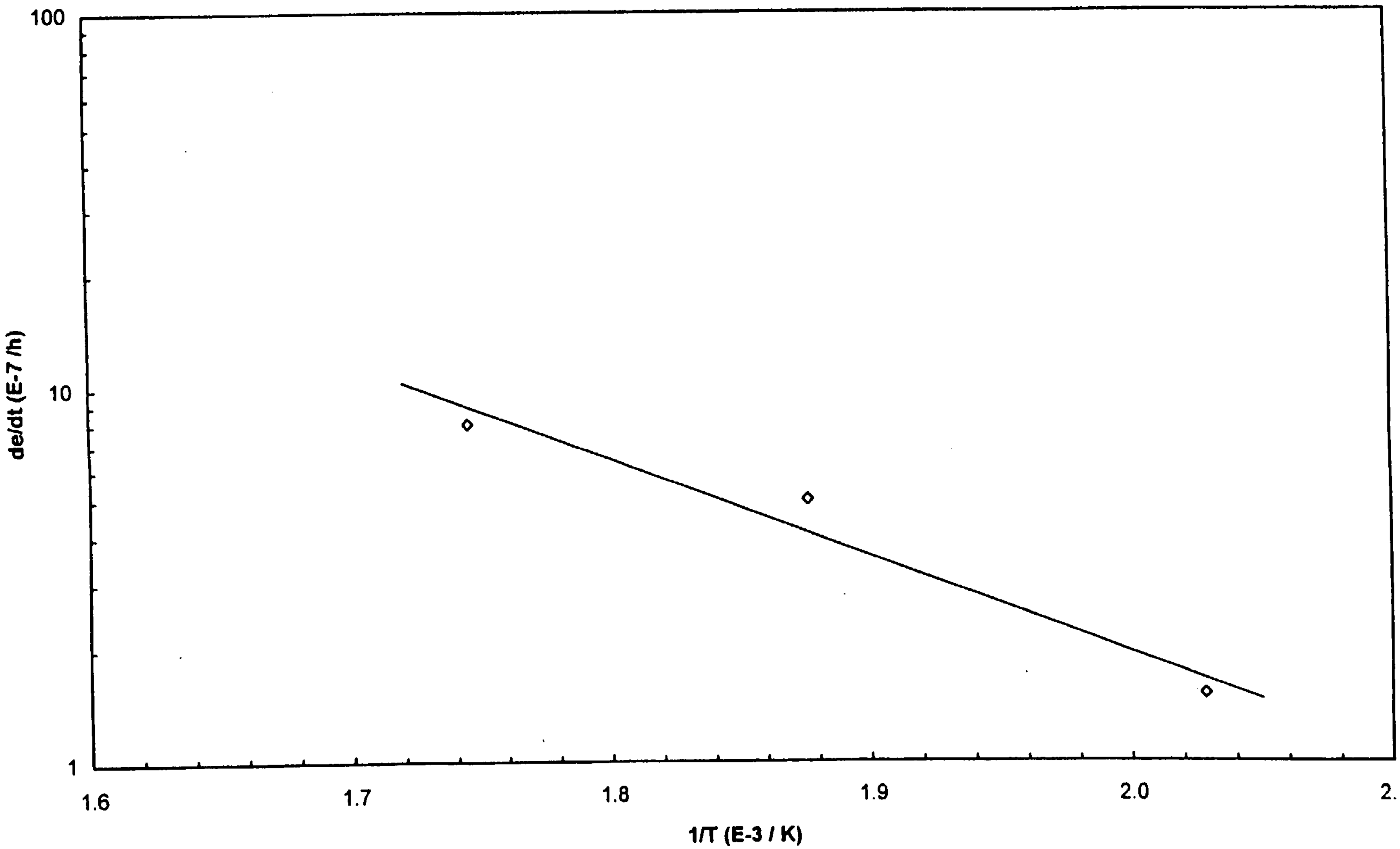
B&W/EPRI, 86 MPa

a,

Figure 16

CW Zr-2 Irradiation Creep Temperature Dependence, 207 MPa

From Reference 16



Graphical representation of data from Reference 16

This figure was not included in Reference 16

Figure 17

CWSR IMP Zr-4 In-Reactor Creep

1.08×10^{22} n/cm^2 (E > 1 MeV), 41 MPa (6.0 ksi) Hoop Stress

a,

Section 2

Other PAD Model Changes

1.0 Revised PAD Code Summary of Changes

The following changes have been incorporated into the revised PAD code:

1) Revised Creep Model (described in Section 1 of this report),
2) Revised Rod Irradiation Growth Model,
3) Updated Zr-4 and ZIRLO™ Clad Thermal Conductivity Values,
4) Updated Zr-oxide Thermal Conductivity Value,
5) Updated Equation of State (EOS) Model,
6) Variable Oxide-Metal Ratio Model (as discussed during the Westinghouse/NRC meeting on May 5, 1998), and
7) Gas Absorption in Cladding Effect.

Item 1 was submitted in the original version of WCAP-15063-P. Items 2 through 7 were presented to the NRC as additional model changes that were being incorporated into the PAD model during a meeting with the NRC and Battelle Northwest Labs (reviewer of the WCAP). These latter items were requested to be incorporated into WCAP-15063-P by the NRC since that had been reviewed along with the revised creep model and would be the basis for the new PAD 4.0 model.

1.1 Revised Creep Model

Description: Refer to WCAP-15063-P for creep model details. This is a substantial improvement in the creep model, which is an important model with wide-reaching impacts for all of the subsequent calculations in the revised PAD code.

Why Change?: The revised creep model is fundamentally sound and has a much more rigorous in-reactor data base to support the mechanistic understanding of in-reactor creep. The new model incorporates temperature-dependent irradiation creep and irradiation hardening.

Effect of change: The overall impact of the revised creep model in the revised PAD code on rod internal pressure predictions is favorable.

1.2 Revised Rod Irradiation Growth Model

Description: The current PAD model (WCAP 10851-P-A) does not have a temperature dependence for irradiation growth of zirconium alloys. The revised PAD model incorporates this temperature dependence.

Why change?: This change is based on work reported by the industry which has been demonstrated at EBR-II and DIDO that irradiation growth is a strong function of temperature, particularly for temperatures above 660 K (728 °F).

Effect of change: This is a relatively small change which will only effect rod growth when high temperatures are present in the cladding.

1.3 Updated Zr-4 and ZIRLO™ Clad Thermal Conductivity Values

Description: PAD (WCAP-10851-P-A) currently uses conductivity values from open literature for Zircaloy-4 and Zircaloy-2, for Westinghouse Zr-4 and ZIRLO™. The revised PAD code uses measured values on Westinghouse fuel products; for both ZIRLO™ and Zr-4.

Why change?: Experimental work was conducted specifically to update the database for Westinghouse product, and when incorporated will substantially improve thermal model accuracy.

Effect of Change: This update has a positive impact on rod internal pressure by slightly lowering clad temperatures for a given power level.

1.4 Updated Zr-oxide Thermal Conductivity Values

Description: PAD (WCAP-10851-P-A) currently uses a value for zirconium-oxide thermal conductivity based on work done in 1979 on theoretically-dense zirconium-oxide in a vacuum. Recent EPRI-sponsored work shows that the oxide thermal conductivity is higher than that currently included in PAD. Oxide thermal conductivity has been revised in the new version of PAD based on this work.

Why Change?: Recent in-pile tests indicate that a more conductive thermal oxide layer is formed in PWR environments, which enhances the oxide thermal conductivity. This change will enable more accurate assessments of the rod thermal response characteristics consistent with industry understanding of zr-oxide properties.

Effect of Change: This change yields a small reduction in clad average temperature and thus a reduction in fuel centerline temperature.

1.5 Equation of State (EOS) Gas Model

Description: PAD (WCAP-10851-P-A) currently uses the ideal gas law for calculating the pressure inside the fuel rods. A review of the available state-of-the art gas laws, show that a new equation of state (EOS) model is more accurate. The revised PAD code uses the Peng-Robinson EOS model for the calculation of fuel rod internal pressure.

Why change?: Changing the PAD gas model from the ideal gas law to the Peng-Robinson EOS will more accurately represent the internal gas pressure of Westinghouse fuel rods.

Effect of Change: This model causes the predicted rod pressure to increase for a given burnup higher than the current ideal gas law and has a small negative effect on rod internal pressure.

1.6 Variable Oxide-Metal Ratio Model

Description: PAD (WCAP-10851-P-A) currently uses a constant theoretical oxide-metal ratio 1.56 to calculate metal wastage. Westinghouse previously identified to the NRC (May 1998) that we would be using a value of [

] a, c.

Why change?: The change in oxide characteristics as the thickness increases has been documented in public literature and measured on archive hot cell photomicrographs.

Effect of Change: This change allows for accurate calculation of remaining wall thickness as oxide is generated and thus improves accuracy of the clad stress and creep calculations.

1.7 Gas Absorption in Cladding Effect

Description: PAD (WCAP-10851-P-A) currently models that air can contribute to the internal pressure of the fuel rod throughout life. Air is rapidly absorbed into the cladding by forming hydrides, oxides and nitrides of zirconium and is eliminated from gas pressure calculations in the revised PAD code.

Why change?: Published literature on diffusion/reaction rates for gases in zirconium alloys, confirms a rapid consumption of any air or reactive gases is expected at operating fuel temperatures. [

] a, c.

Effect of Change: This change will result in a small reduction in rod internal pressure.

2.0 Revised Rod Irradiation Growth Model

2.1 Model Background and Justification

Extensive in-reactor testing has been performed in EBR-II (fast neutron spectrum) and DIDO (thermal spectrum). One set of tests reported by Rogerson determined the irradiation growth in DIDO and EBR-II with the same material (RXA Zr-2)(1). The result shows that the growth strain exhibited by the EBR-II sample is within the sample-to-sample scatter exhibited by the DIDO data. This result shows that irradiation growth data measured in a fast neutron spectrum (specifically EBR-II) is applicable to thermal neutron spectra. Therefore, the EBR-II data may be applied to PWRs.

The available CW irradiation growth data covers an extensive parameter range. The temperatures are in the range of 353 to 687 K (176 to 777 °F), and the fluences extend up to values similar or higher than typical end-of-life PWR fuel rods (1.7×10^{22} n/cm^2, E > 1 MeV). In the case of the EBR-II tests, large growth strains were observed (strains as large as 2.5%)(2). At high fluences (> 0.5×10^{22} n/cm^2, E > 1 MeV) and temperatures > 650 K, the irradiation growth strain and strain rate is the same for CW and RXA material. Figures from Reference 2 for 20% CWSR Zr-2 slab material show that this behavior is not texture or temperature dependent (for temperatures > 650 K).

2.2 PAD Revision

The revised PAD irradiation growth equation was modified using the irradiation growth rate temperature dependence reported by Fidleris et. al. (3). At temperatures > 660 K, the irradiation growth rate increases rapidly with increasing temperature. The high temperature effect (for temperatures > 660 K), was modeled by [

] a, c (see Figure 2-1):

$$[\ G/G_o = 0.0212\,T(K) - 12.967 \qquad for\ T > 660\ K\ ]^{a,c}$$

[

] a, c.

Figure 2-1

G/G_0 versus Temperature

a,c

3.0 Updated Zr-4 and ZIRLO™ Clad Thermal Conductivity Values

3.1 Model Background and Justification

Table 3-1 summarizes the thermal conductivity values calculated as a function of temperature, based on the tests conducted by the "Properties Research Laboratory" in West Lafayette, Indiana, on Westinghouse Zircaloy-4 and ZIRLO™ cladding. [

] a, b, c.

A linear fit of the data presented in Table 3-1 yields the following best estimate model for Zircaloy-4, in the temperature range of [ ] a, b, c.

[ ] a, b, c (1)

where, k = Thermal Conductivity in $Wcm^{-1}K^{-1}$
T = Temperature in °C

In the case of ZIRLO™, a linear fit of the data presented in Table 3-1 yields the following best estimate model in the temperature range of [ ] a, b, c.

[ ] a, b, c (2)

where, k = Thermal Conductivity in $Wcm^{-1}K^{-1}$, and
T = Temperature in °C

Figure 3-1 represents the linear plots of thermal conductivity versus temperature for Zircaloy-4 and ZIRLO™ as represented by Equations 1 and 2 respectively.

3.2 PAD Revision

In view of the fact that the maximum allowable clad design temperature for steady state operation for Zircaloy-4 is [ ] a, c and for ZIRLO™ is [ ] a, c respectively, and for Condition II transients is [ ] a, c for Zircaloy-4 and [ ] a, c for ZIRLO™, models represented by Equation (1) for Zircaloy-4 and Equation (2) for ZIRLO™ clad will be used for thermal conductivity predictions as a function of temperature for Westinghouse fuel clad in the revised PAD code.

Table 3-1

Thermal Conductivity as a function of temperature

a, c

[redacted]

Figure 3-1
Thermal Conductivity as a Function of Temperature for Zircaloy-4 and ZIRLO™ (1st Order Fit)

a, c

4.0 Updated Zr-Oxide Thermal Conductivity Value

4.1 Model Background and Justification

A best estimate value of [ ] a, b, c has been used in PAD (WCAP-8720) for Zr-4 oxide layers, based on data from Reference 4. Since that time, additional data have become available (References 5 and 6) which indicates that the oxide layer thermal conductivity has a higher value. The purpose of this update is to use the appropriate data from References 5 and 6 to establish a revised best-estimate average value of the Zr oxide layer thermal conductivity.

The first set of new ramp tests (Reference 5) consisted of a set of 12 ramps on four rods with oxide layer thicknesses of 30, 54, 66 and 82 microns. Thermal conductivity values ranged from 1.4 to 3.7 W/mK with an average value of 2.4 W/mK.

The second set of ramp tests, Reference 6, was run because there was no clear dependence of oxide thermal conductivity on oxide layer thickness, and it seemed that crud could have been present on two of the rods and that could have affected the results. The two fuel rods were brushed and the tests were repeated.

During the second set of ramps, it was noted that the rod elongation during up-ramps was greater than contraction during down-ramps. It was postulated that pellet-cladding mechanical interaction could be occurring during the up-ramps, and it was recommended that only the down-ramps be used for thermal conductivity measurements. The oxide layers were re-measured, and it was found that the thickness of the 26 micron layer was actually 30 microns.

A total of seven down ramps were measured during the second set of experiments. The resulting thermal conductivity values are given in Table 3-2 of Reference 6. A summary of the thermal conductivities from the first set of ramps is given in Table 4-1.

Table 4-1

Summary of Conductivity Values From First Set of Ramps

a, b, c

These data were combined with data from the second set of ramps given in Table 3-1 of Reference 6. The combined data are given in Table 4-2. The variables are the TEST series, 1 or 2, the RAMP number, UP1D2 = 1 for up ramps and 2 for down ramps, t = the oxide thickness, and K = the thermal conductivity.

Table 4-2

Summary of Conductivity Data

a, b, c

Table 4-2 (cont.)

Summary of Conductivity Data

a, b, c

Table 4-2 (cont.)

Summary of Conductivity Data

a, b, c

[ ]

The results of a statistical analysis of the Table 4-2 data for the down ramps are given below:

[ ] a, c

[ ] a, c

[ ] a, c

Tests were performed to determine if the conductivity values are normally distributed. [ ] a, c.

4.2 PAD Revision

In conclusion, the NFIR conductivity data can be characterized and included in the revised PAD code as follows:

[ ] a, c

for evaluating cladding temperatures with oxide layers present.

5.0 Updated Equation of State Model (EOS)

5.1 Model Background and Justification

The relationship between pressure, temperature, and mass for a fission gas in PAD (WCAP-10851-P-A) is based on the Ideal Gas Law,

$$PV = NRT \tag{1}$$

where P and T and the pressure and temperature of the gas respectively, V is the volume occupied by the gas, and N is the number of moles of gas. The universal gas constant is R. Equation (1) is equivalent to:

$$Pv = RT \tag{2}$$

where v is the specific volume and R is a (particular) gas constant.

The Ideal Gas Law relationship is valid for many gases near room temperature and pressure, and is good for noble gases such as helium, neon, and argon up to moderate pressures (400 - 500 psia). At high pressures ($P > 500$ psia) however, the Ideal Gas Law becomes increasingly inaccurate. Figure 1 from Reference 8, shows the compressibility of several gases at high pressure, where compressibility Z is defined as:

$$Z = \frac{Pv}{RT} \tag{3}$$

For an ideal gas, the compressibility $Z = 1.0$. Deviation from 1.0 is an indication of non-ideal behavior, and that the use of the Ideal Gas Law will lead to inaccurate results.

Figure 1 from Reference 8, shows that above about 500 psia, inert gases do not exhibit ideal behavior.

At end of life, when the rod internal pressure can exceed 2000 psia, none of these gases will exhibit ideal behavior. Therefore, use of the Ideal Gas Law to estimate rod pressure given the temperature and specific volume will be inaccurate. Since helium makes up the majority of the gas, and the compressibility of helium is greater than 1.0, the Ideal Gas Law will underpredict the actual rod pressure.

A survey was conducted to determine the most appropriate EOS. It was determined that the Peng-Robinson equation, Reference 9, gave the most accurate predictions for the range of interest. For fission gas mixtures even at end of life, helium has the highest mol fraction. To check these EOSs for high He mol fraction, the

results were plotted as DP/P versus helium mol fraction. These results are shown in Figures 5-1 and 5-2. Measured data was obtained from References 10 through 14, for the various gas matrix evaluations.

Note that DP/P is defined as:

$$\frac{DP}{P} = \frac{P_{data} - P_{prediction}}{P_{data}} \tag{4}$$

This quantity is positive when the pressure is underpredicted, and negative when overpredicted.

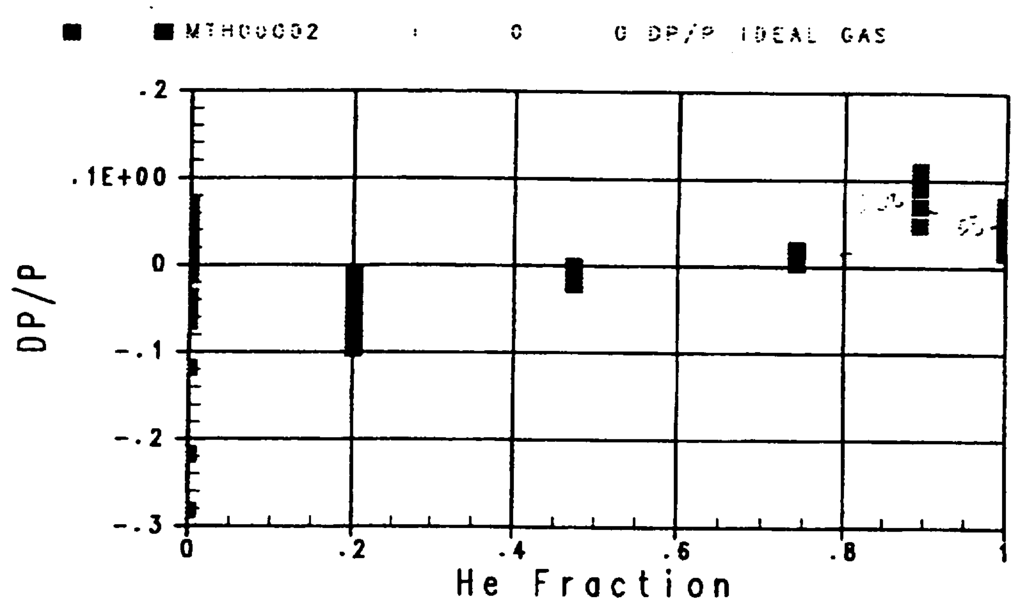


Figure 5-1

Ideal Gas Equation of State

DP/P versus He Mol Fraction

From Figure 5-1 it becomes clear that for gas compositions rich in helium, the Ideal Gas Law may underpredict the actual pressures, in some cases by more than 10%.

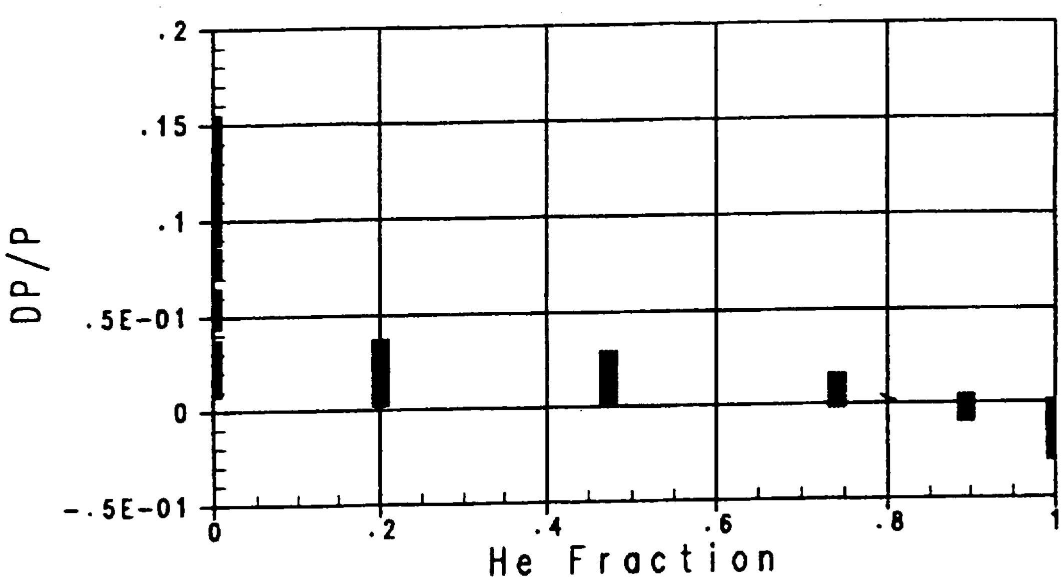


Figure 5-2

Peng-Robinson Equation of State

DP/P versus He Mol Fraction

The Peng-Robinson EOSs performs well at high He composition. For mol fractions greater than about 0.2, it predicts the pressure to within +/- 5%. At mol fractions approaching 1.0, the Peng-Robinson EOS overpredicts the pressure slightly, but never by more than 3% as shown in Figure 5-2.

Figure 5-3 shows the Peng-Robinson M-P plot results for various pressures. The pressure predictions for mixtures with high helium mol fractions are always predicted to within 5%.

5.2 Summary

The Ideal Gas Law was found to potentially underpredict pressure for compositions with high helium mol fraction. The more complex cubic Equations of State (Redlich-Kwong, Soave, and Peng-Robinson) were more accurate at high helium mol fraction, and tended to overpredict the pressure slightly. Of these three, the Peng-Robinson EOS was found to be slightly better than the other two.

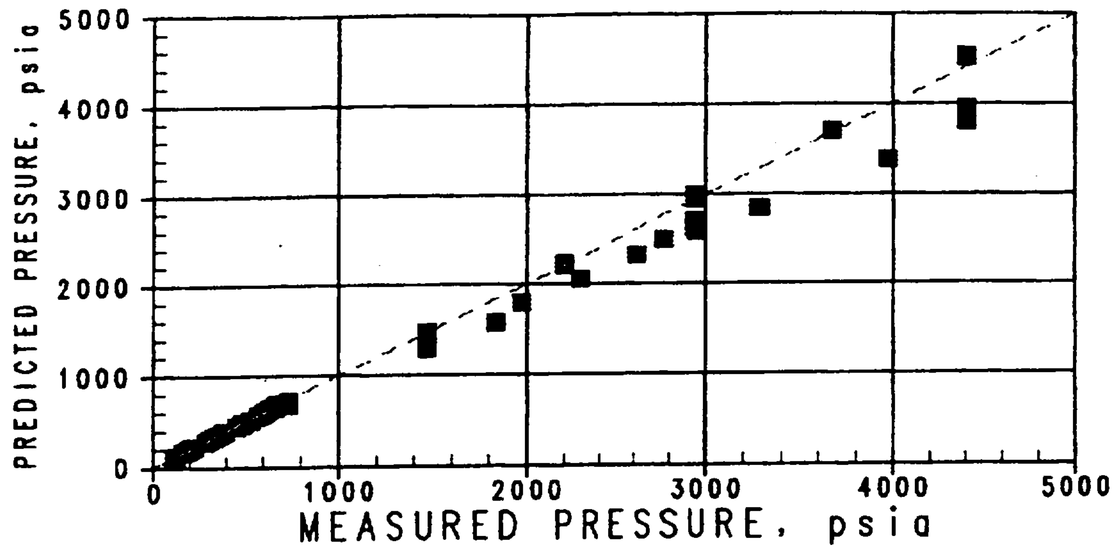


Figure 5-3
Peng-Robinson Equation of State
Measured versus Predicted Pressure

5.3 Pad Revision

PAD revision investigations of several Equations of State applicable to gas mixtures when compared to available data over a range of pressure, temperature, and composition shows the Peng-Robinson EOS is most accurate and will be used in the revised PAD code.

The pressure-temperature-volume relationship for a pure fluid is often represented by a cubic Equation of State, which has the general form:

$$P = \frac{RT}{v - b} - \frac{a}{v^2 + ubv + wb^2} \tag{5}$$

where P is the pressure, T is temperature, v is specific volume, and R is the Universal Gas constant.

For the Peng-Robinson equation of state. $u = 2.w = -1$ with

$$b = \frac{0.07780 RT_c}{P_c} \tag{6}$$

and

$$a = \frac{0.45724 r^2 T_c^2}{P_c}[1 + f\omega(1 - T_r^{0.5})]^2 \tag{7}$$

where,

$$f\omega = 0.37464 + 1.54226\omega - 0.26992\omega^2 \tag{8}$$

In Equations (6) and (7), the subscript "c" denotes properties at the critical point. The reduced pressure is defined as

$$T_r = \frac{T}{T_c} \tag{9}$$

The function for $f\omega$ given by Equation (8) uses the acentric factor ω, which is a parameter that represents the complexity of a molecule with respect to geometry and polarity. For mono-atomic gases, ω is usually zero or very small.

In the revised PAD code up to seven different gases can be present in the gas mixture. The following table lists these components and the properties assigned in the code as taken from Reference 9:

Table 5-1
Peng-Robinson Equation of State
Component Properties List for PAD

Component	Tcrit. K	Pcrit, bar	ω
Helium	5.19	2.27	-0.365
Xenon	289.7	58.4	+0.008
Krypton	209.4	55.0	+0.005
Argon	150.8	48.7	+0.001
Nitrogen	126.2	33.9	+0.039
Water Vapor	647.3	221.2	+0.344
Hydrogen	33.2	13.0	-0.281

For gas mixtures, the attraction and repulsion between molecules of different components causes non-linear variation of some properties with composition. To account for this in Equation (5), a set of mixing rules can be defined to this non-linearity. The values of "a" and "b" in Equation (5) are re-defined. Based on the recommendations in Reference 4, the following mixing rules are used in the revised PAD code.

$$a_m = \sum_i \sum_j y_i y_j (a_i a_j)^{0.5} (1 - k_{ij}) \tag{10}$$

$$b_m = \sum_i y_i b_i \tag{11}$$

The b_i and a_i for each pure component are given by Equations (6) and (7) respectively. The term k_{ij} is used for some binary pairs to adjust for strong interactions and is determined from experimental data. In the revised PAD code $k_{ij} = 0$, is assumed for all binary combinations.

6.0 Variable O-M Ratio Model

6.1 Model Background and Justification

In order to accurately model fuel rod clad temperatures and stresses in a fuel performance models as well as 17% metal wastage calculations, an accurate model of Zircaloy-4 oxide to metal ratio is needed for use in design. Due to the differences in densities of the oxide and the base metal, there is a volumetric change from the metal consumed to the oxide formation. This volumetric difference results in a thicker oxide than the metal that was consumed. The ratio of the volumes is characterized by the oxide-to-metal ratio (O/M). The theoretical oxide-to-metal ratio is referred to as the Pilling-Bedworth ratio, and for Zirconium based alloys the value of 1.56 is commonly used. However, during the in-reactor generation of ZrO_2, different mechanisms occur that cause the oxide density to be less than theoretical resulting in higher O/M ratios at increasing oxide thicknesses. Westinghouse metallographic O/M measurements from fuel rod hot cell programs were evaluated and a predictive model was generated which relates O/M with oxide thickness. [

] $^{a, c}$. This model was first presented to the NRC by presentation on May 5, 1998.

6.2 Variable O/M Ratio Model Details

As the oxide grows, it transitions from a protective to a non-protective structure. The non-protective oxide contains cracks and pores and this transition occurs when the oxide is about [ ] $^{a, c}$. In generating a model which predicts O/M ratio as a function of oxide thickness, the first [ ] $^{a, c}$ of oxide should result in a constant theoretical value of O/M ratio. At higher oxide thicknesses, the data presented in the previous section is used to develop the relationship of O/M ratio with increasing oxide thickness.

The equations governing the O/M ratio as a function of oxide thickness are as follows:

$$\frac{O}{M} = \frac{O}{M_{Th}}, \quad t < [\quad]^{a,c} \tag{1}$$

$$\frac{O}{M} = [\qquad\qquad\qquad]^{a,c} \tag{2}$$

where: O/M_{th} = theoretical value of O/M ratio = 1.56

a = fitting coefficient, [ ] a, c

b = fitting coefficient

t = oxide thickness (mils)

Equations 1 and 2 are combined and plotted against the sorted data from O/M measurements in Figure 6-1.

Figure 6-1

Best Estimate O/M Ratio Model

a, b, c

7.0 Gas Absorption in Cladding Effect

7.1 Model Background and Justification

The fuel rod internal gas mixture includes: (1) the fission gasses produced during operation, (2) gas from the pellets including the gas from the IFBA coating if present, (3) the gas from the rod pre-pressurization, and (4) the [ ]^{a,c}. When the rod is pre-pressurized and sealed during fabrication, the rod [ ]^{a,c}. Both IFBA and non-IFBA rods contain this equivalent volume [ ]^{a,c}. Zirconium alloys are known to react with [ ]^{a,c}.

For example, assuming a plenum volume of about [ ]^{a,c} and a gas mixture of [ ]^{a,c}.

With about [ ]^{a,c} the corresponding weight gain for total [ ]^{a,c}. Based on reaction rates in Reference 16, [ ]^{a,c} will occur within [ ]^{a,c}; thus, all of the [ ]^{a,c}.

Zirconium preferentially reacts with [ ]^{a,c} are present. The reaction rate with [ ]$^{a,c(2)}$. When the [ ]^{a,c}. The absorption rate of [ ]^{a,c}. Based upon the weight of [ ]^{a,c}. Thus, it will take about [ ]^{a,c}. This may be a lower than actual rate since rate is temperature dependent. [ ]^{a,c}.

Irradiated rods were punctured in the hot cell and the gas present in the rod was captured and analyzed. In the 22, [ ]^{a,c} measured and in 7 other rods from [ ]^{a,c} there was [ ]^{a,b,c}. The measurement sensitivity is reported as less than 0.01% by volume. If the [

] a, c pressurization and a resultant internal rod gas mixture with [ ] a, c by volume. These levels, if present in the rod at end of life, are above the detection limit by a factors of over 100.

7.2 PAD Change

Evaluations indicate that the [ ] a, c the Zirc-4 or ZIRLO™ microstructure and [ ] a, c. Based on this evaluation, the revised PAD code will not use [ ] a, c.

8.0 References

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Attachment 2

Proprietary and Non-Proprietary Versions

Of

WCAP-15063-P Errata for Section 1

Errata for WCAP-15063-P, Revision 1, Section 1

"Westinghouse In-Reactor Creep Model"

Based on the responses to the Requests for Additional Information (RAIs) on WCAP-15063-P, "Westinghouse In-Reactor Creep Model", that were received from the NRC on September 10, 1998, the following is a list of errata revisions to WCAP-15063-P, Revision 1, Section 1. The text to be inserted and to be deleted are enclosed by quotation marks.

1. Coefficient C_3:

An editorial change is necessary to be made to WCAP-15063-P, Revision 1, Section 1, in order that the method used to evaluate the coefficient C_3 is clearly presented. The evaluation of C_3 is correctly described in the last paragraph of Page 14 but not correctly described in the middle of Page 6. The editorial change involves the deletion of one sentence and the insertion of several sentences on Page 6. The sentence to be deleted on Page 6 is:

"The equation was [ ]^{a,c} IMP Zr-4 out-of-reactor (laboratory) thermal creep tests [ ]^{a,c} and the out-of-reactor (laboratory) creep rate equation for [ ]^{a,c} according to:

[ ]^{a,c} (9)"

The sentences to be inserted in place of the deleted sentence are:

"The equation was [ ]^{a,c} out-of-reactor (laboratory) thermal creep rates [ ]^{a,c}. The calculation was performed for typical fuel rod parameters. The parameters used were [ ]^{a,c}. C_3 was evaluated according to:

[-]^{a,c} (9)

The results of the analysis are listed in Table 7."

Table 7

Calculated values of the coefficient C_3

T (K)	C_3
589	[ ]^{a,c}
616	[ ]^{a,c}
644	[ ]^{a,c}
672	[ ]^{a,c}
avg. =	[ ]^{a,c}

2. **Irradiation Hardening:**

An evaluation of the diameter strain exhibited by a cold-worked (CW) Zr-2 pressure tube shows that the Zr-2.5Nb data are applicable to Zr-4. When WCAP-15063-P was prepared, the CW Zr-2 pressure tube data were not known. The availability of the CW Zr-2 pressure tube strain data strengthens the conclusion that the Zr-2.5Nb data are applicable to Zr-4. Section 3.2.1.2 in WCAP-15063-P, Revision 1, Section 1, presents [

]^{a,c}. The CW Zr-2 pressure tube may be added to WCAP-15063-P, Revision 1, Section 1, by modifying one sentence on Page 11 in the first paragraph of Section 3.2.1.2, adding text just after the modified sentence, modifying Table 3, adding Figure 18, and adding Reference 25. The sentence to be modified is:

"[

]^{a,c}"

The modified sentence is:

"[

]^{a,c}"

The text to be added is:

"[

]^{a,c}. This shows that the Zircaloy-2.5Nb data are applicable to Zircaloy-4."

Reference:

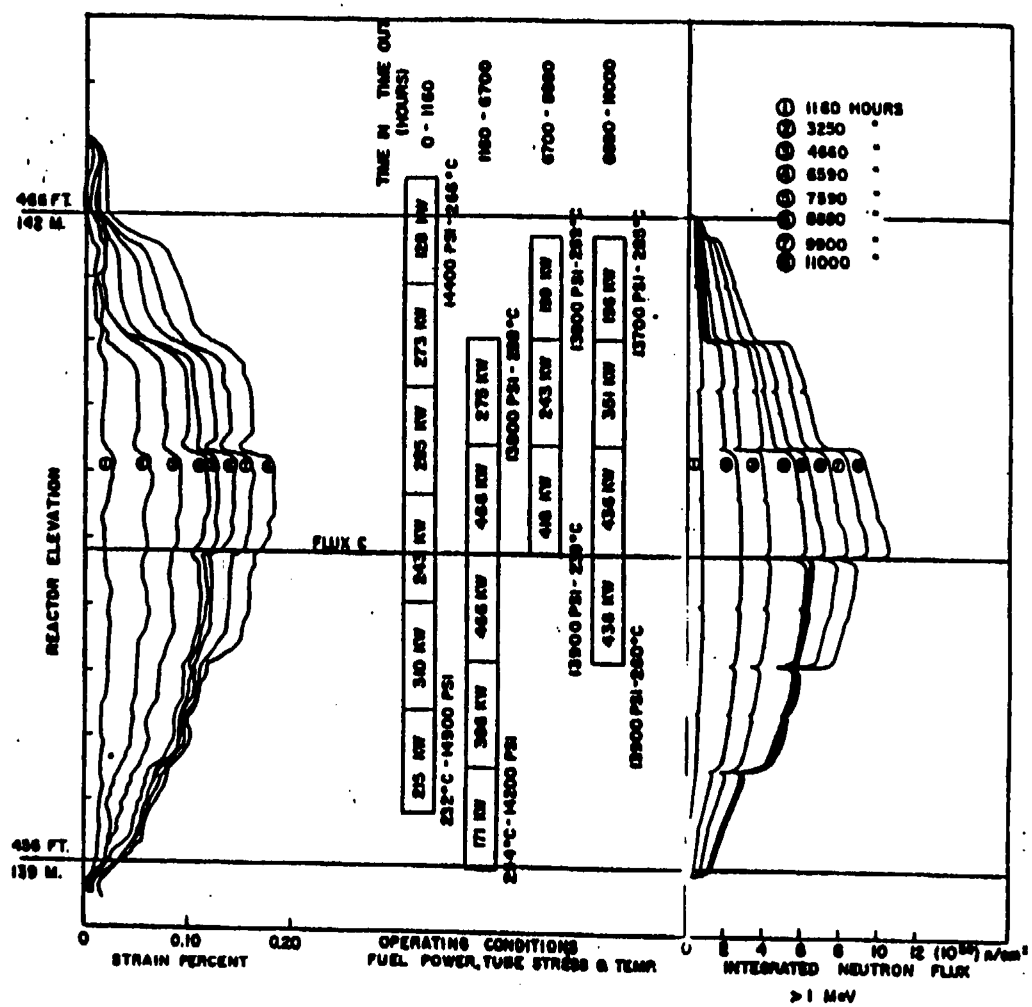
25. P. A. Ross-Ross and C. E. L. Hunt, "The In-Reactor Creep of Cold-Worked Zircaloy-2 and Zirconium-2.5 wt % Niobium Pressure Tubes," Journal of Nuclear Material, Volume 26, 1968, pages 2 through 17.

Table 3

Irradiation Hardening Reduction Factors Evaluated by Strain Ratio and Strain Rate Ratio Analysis

a, c

Figure 18



Reprinted from P. A. Ross-Ross and C. E. L. Hunt, "The In-Reactor Creep of Cold-Worked Zircaloy-2 and Zirconium-2.5 wt % Niobium Pressure Tubes," Journal of Nuclear Materials, Volume 26, 1968, pages 2 through 17, with permission from Elsevier Science

3. **Typographical Errors:**

- The date of reference 12 is incorrect. Change the date for Reference 12 from "1974" to "1993".

- The saturated primary strain in Equation 14 does not include the superscript. For the left hand side of Equation 14, change " e_p " to " e_p^{sat} ".

4. **Thermal Creep Equations:**

The in-reactor thermal creep equations presented in WCAP-15063-P represent asymptotic approximations. These approximations were used for simplicity. After preparation of WCAP-15063-P, the equations were coded into PAD, and the exact equations were used. Therefore, the exact equations used by PAD need to be included in WCAP-15063-P. This may be accomplished by completely replacing section 2.1 with the following.

"2.1 In-Rector Thermal Creep Overview

The new in-reactor creep model was developed to describe Westinghouse cold-worked stress relieved (CWSR) improved (IMP) Zr-4 (low tin Zr-4) tubing. The model is based on [ ]^{a,c}. The in-reactor thermal creep is given by the out-of-reactor (laboratory) thermal creep corrected for in-reactor irradiation hardening. This behavior is described by:

[ ]^{a,c} (3)

where [ ]^{a,c}. The equation [ ]^{a,c}. The equation for [ ]^{a,c} is given by:

[ ]^{a,c} (11)

where [ ]^{a,c}. The thermal creep strain given by [ ]^{a,c} IMP Zr-4 thermal creep tests [ ]^{a,c} according to:

[ ]^{a,c} (4)

The expression for IH is:

[ ] a, c (5)

where [ ] a, c. Equation (5) provides a smooth transition with increasing fluence from no irradiation hardening [ ] a, c to complete irradiation hardening [ ] a, c. The irradiation hardening [ ] a, c. The application of the [ ] a, c.

Since both [ ] a, c Thus:

[ ] a, c (6)

The derivative of the numerator is given by:

[ ] a, c (35)

The derivative of the denominator is given by:

[ ] a, c (36)

and the expression for de/dt(tc) is:

[ ] a, c (37)"

5. Correlation of In-Reactor with Out-Reactor Creep:

The out-of-reactor (laboratory) thermal creep rates are directly related to the in-reactor irradiation creep rates for a given zirconium alloy. This relationship [ ] a, c. WCAP-15063-P presents the analysis of the [ ] a, c. The Westinghouse BR-3 ZIRLO™ fuel rod data and Westinghouse North Anna ZIRLO™ fuel rod data may be included by deleting one sentence and adding two paragraphs. The sentence to be deleted is the last sentence of the 2nd paragraph on Page 14:

"[

]^{a,c}."

The following two paragraphs are to be added at the beginning of Section 3.2.2.2, Normalization of B&W/EPRI Irradiation Creep to Westinghouse Behavior:

"The out-of-reactor (laboratory) thermal creep rates are directly related to the in-reactor irradiation creep rates for a given zirconium alloy. This relationship [

]^{a,c}.

In the case of the [

]^{a,c}."

Table 8

ZIRLO™ North Anna In-Reactor Creepdown

and Out-of-Reactor Thermal Creepout

a, c

Figure 19

Westinghouse BR-3 Fuel Rod Profilometry (Creepdown)

ZIRLO™ 15x15 Cladding

a, c

Figure 20

Out-of-Reactor Thermal Creep (Creepout)

658 K (725 °F), 108 MPa (15.6 ksi) Equivalent Stress

a, c

Figure 21

Comparison of In-Reactor and Out-of-Reactor Creep Rates

Westinghouse ZIRLO™

[ ]a

6. Creep Model Predictions:

The text describing the predictions of the in-reactor creep model were inadvertently omitted. The following section should be added:

"3.3 In-Reactor Creep Model

The behavior of the in-reactor creep model is illustrated in Figure 17. The calculation of the creep rates were performed for typical fuel rod parameters. The parameters were [

][a,c]. The out-of-reactor thermal and in-reactor thermal creep components are also shown."

Errata for WCAP-15063-P, Revision 1, Section 2
"Other PAD Model Changes"

1. **Typographical Error:**

- The number of down ramps in the last paragraph on page 53 is incorrect. Change the number of down ramps from "seven" to "six".

2. Updates were made to the Equation of State section of the WCAP due to errors that were found in the original package. Replace Section 5.0, "Updated Equation of State Model (EOS)" in its entirety. Please note that the Table of content and figure titles will be revised accordingly.

5.0 Updated Equation of State Model (EOS)

5.1 Model Background and Justification

The relationship between pressure, temperature, and mass for a fission gas in PAD (WCAP-10851-P-A) is based on the Ideal Gas Law,

$$PV = NRT \tag{1}$$

where P and T and the pressure and temperature of the gas respectively, V is the volume occupied by the gas, and N is the number of moles of gas. The universal gas constant is R. Equation (1) is equivalent to:

$$Pv = RT \tag{2}$$

where v is the specific volume and R is a (particular) gas constant.

The Ideal Gas Law relationship is valid for many gases near room temperature and pressure, and is good for noble gases such as helium, neon, and argon up to moderate pressures (400 - 500 psia). At high pressures (P > 500 psia) however, the Ideal Gas Law becomes increasingly inaccurate. Figure 5.0, taken from Reference 8, shows the compressibility of several gases at high pressure, where compressibility Z is defined as:

$$Z = \frac{Pv}{RT} \tag{3}$$

For an ideal gas, the compressibility Z = 1.0. Deviation from 1.0 is an indication of non-ideal behavior, and that the use of the Ideal Gas Law will lead to inaccurate results.

Figure 5.0 shows that above about 500 psia, inert gases do not exhibit ideal behavior.

At end of life, when the rod internal pressure can exceed 2000 psia, none of these gases will exhibit ideal behavior. Therefore, use of the Ideal Gas Law to estimate rod pressure given the temperature and specific volume will be inaccurate. Since helium makes up the majority of the gas, and the compressibility of helium is greater than 1.0, the Ideal Gas Law will underpredict the actual rod pressure.

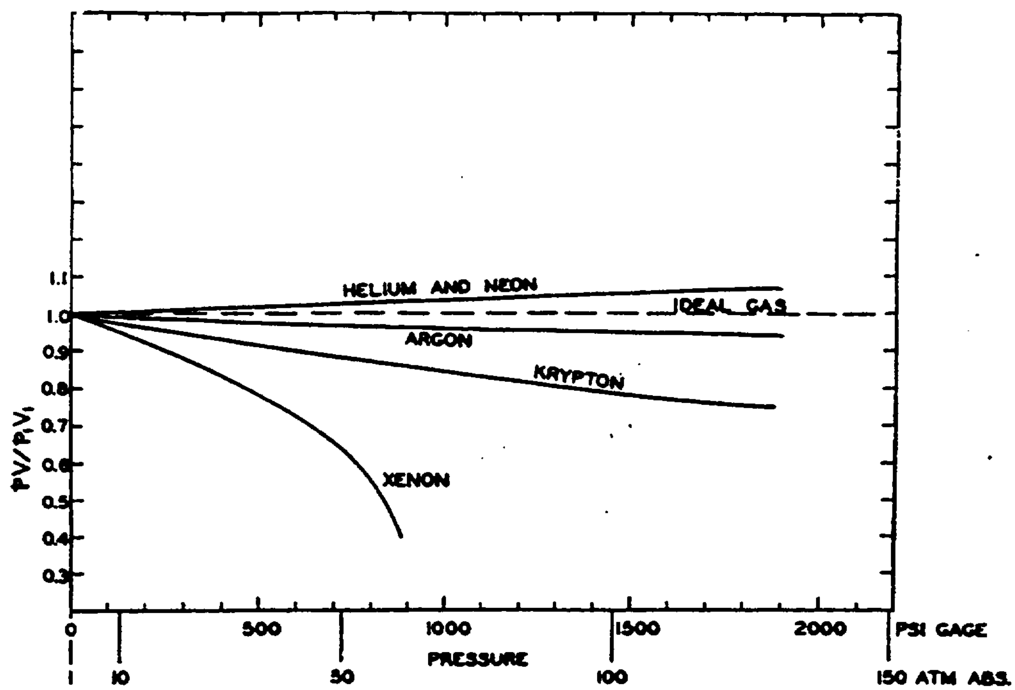


Figure 5-0
Compressibility of Inert Gas at 21°C

A survey was conducted to determine the most appropriate gas EOS. It was determined that the Peng-Robinson equation, Reference 9, gave the most accurate predictions for the range of interest. However, this EOS consistently under predicts the measurements over the whole range of helium mole fraction. A positive aspect of this EOS is that the prediction error is insensitive to the helium concentration, thus leading to the following (calibrated) gas EOS:

$$P^c = \alpha \times P \tag{4}$$

where P^c is the Corrected or Calibrated Peng-Robinson gas EOS, P is the original Peng-Robinson gas EOS and [ ]^{a,c} is a correction factor determined from the data depicted in Figure 5-2. Figures 5-2 show the graph for the Corrected Peng-Robinson EOS. The prediction errors are very balanced and range between -2% and 2%.

For fission gas mixtures even at end of life, helium has the highest mole fraction. To check the EOS for high He mole fraction, the results were plotted as $\Delta P/P$ versus helium mole fraction. These results are shown in Figures 5-1 and 5-2. Measured data was obtained from References 10 through 14, for the various gas matrix evaluations.

Note that $\Delta P/P$ is defined as:

$$\frac{\Delta P}{P} = \frac{P_{prediction} - P_{measured}}{P_{measured}} \tag{5}$$

This quantity is positive when the pressure is over predicted, and negative when under predicted.

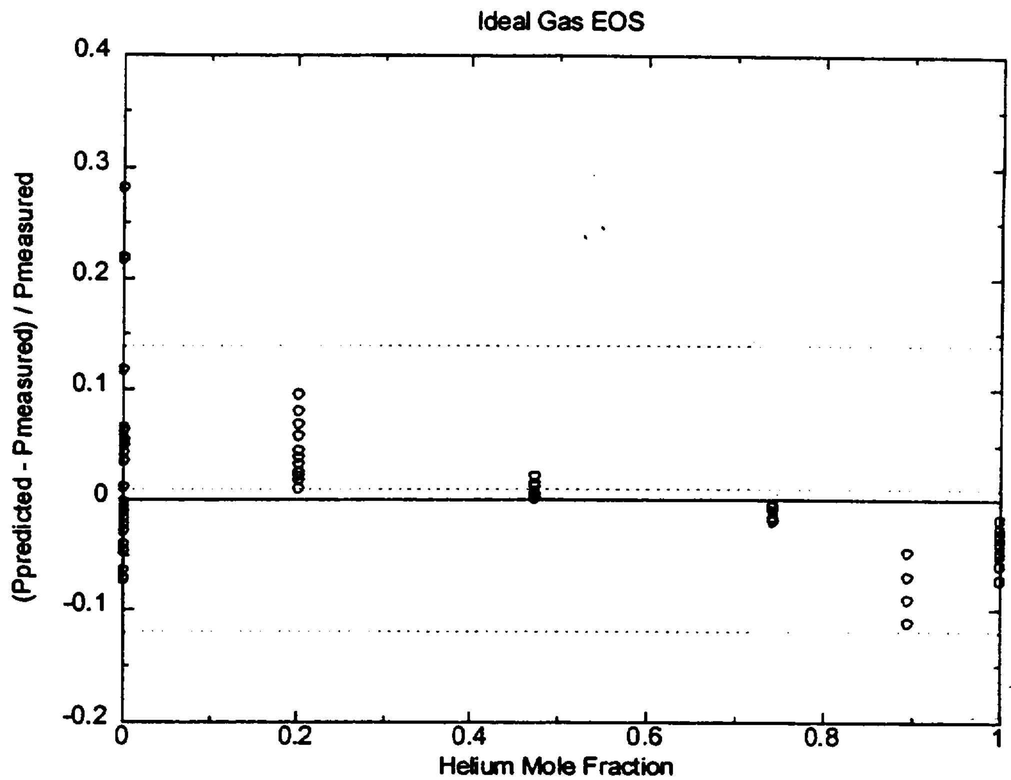


Figure 5-1
Ideal Gas Equation of State
ΔP/P versus Helium Mole Fraction

From Figure 5-1 it becomes clear that for gas compositions rich in helium, the Ideal Gas Law may under predict the actual pressures, in some cases by more than 10%.

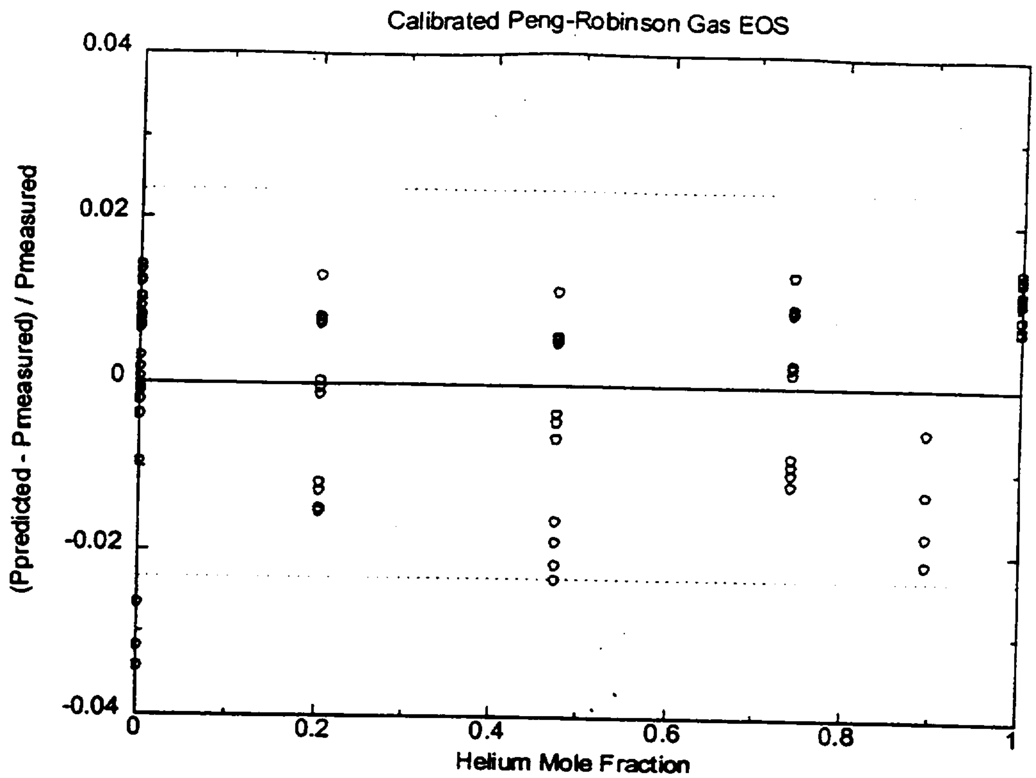


Figure 5-2
Calibrated Peng-Robinson Equation of State
ΔP/P versus Helium Mole Fraction

The Calibrated Peng-Robinson EOS performs well at high He composition. For any mole fraction, it predicts the pressure within +/- 2%. Figure 5-3 shows Calibrated the Peng-Robinson Predicted vs. Measured plot results for various pressures.

5.2 Summary

The Ideal Gas Law was found to potentially under predict pressure for compositions with high helium mole fraction. The more complex cubic Equations of State (Redlich-Kwong, Soave, and Peng-Robinson) were more accurate at high helium mole fraction, and tended to over predict the pressure slightly. Of these three, the Peng-Robinson EOS was found to be slightly better than the other two.

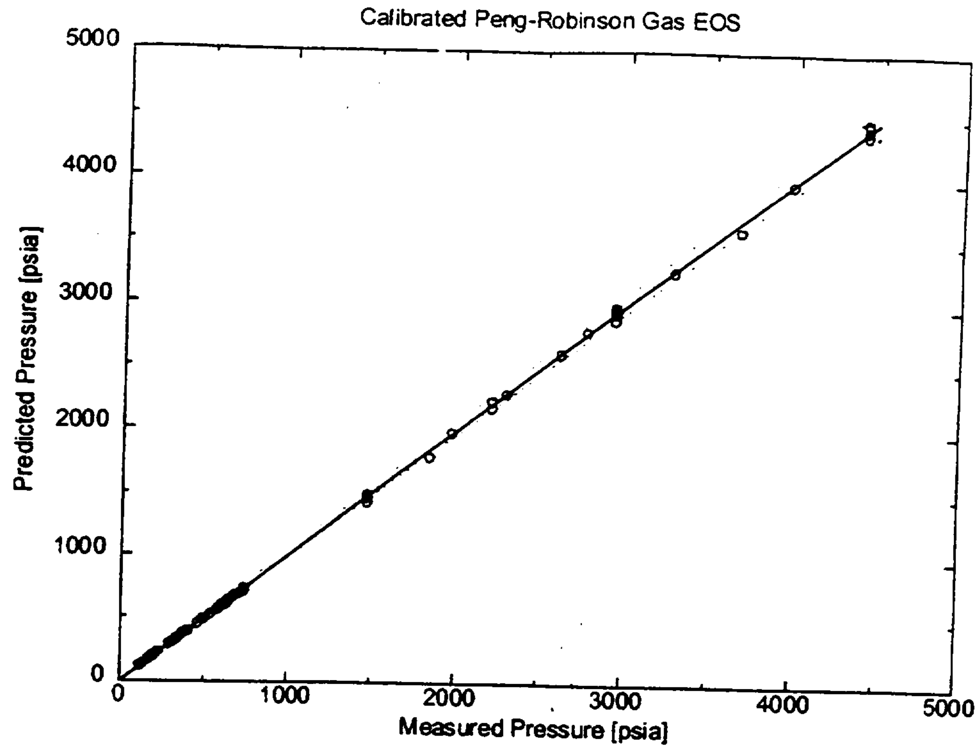


Figure 5-3
Calibrated Peng-Robinson Equation of State
Measured versus Predicted Pressure

5.3 Pad Revision

PAD revision investigations of several Equations of State applicable to gas mixtures when compared to available data over a range of pressure, temperature, and composition shows the Calibrated Peng-Robinson EOS is most accurate and will be used in the revised PAD code.

The pressure-temperature-volume relationship for a pure fluid is often represented by a cubic Equation of State, which has the general form:

$$P = \frac{RT}{v - b} - \frac{a}{v^2 + ubv + wb^2} \qquad (6)$$

where P is the pressure, T is temperature, is specific volume, and R is the Universal Gas constant.

For the Peng-Robinson equation of state. u = 2,.w = -1 with

$$b = \frac{0.07780RT_c}{P_c} \qquad (7)$$

and

$$a = \frac{0.45724\, r^2 T_c^2}{P_c}\left[\ 1 + f\omega(1 - T_r^{0.5})\ \right]^2 \tag{8}$$

where,

$$f\omega = 0.37464 + 1.54226\omega - 0.26992\,\omega^2 \tag{9}$$

In Equations (7) and (8), the subscript "c" denotes properties at the critical point. The *reduced* temperature is defined as

$$T_r = \frac{T}{T_c} \tag{10}$$

The function for given by Equation (9) uses the eccentric factor , which is a parameter that represents the complexity of a molecule with respect to geometry and polarity. For mono-atomic gases, is usually zero or very small.

In the revised PAD code up to seven different gases can be present in the gas mixture. The following table lists these components and the properties assigned in the code as taken from Reference 9:

Table 5-1
Pure Gas Component Properties List for PAD

Component	T_c °K	P_c bar	
Helium	5.19	2.27	-0.365
Xenon	289.7	58.4	+0.008
Krypton	209.4	55.0	+0.005
Argon	150.8	48.7	+0.001
Nitrogen	126.2	33.9	+0.039
Water Vapor	647.3	221.2	+0.344
Hydrogen	33.2	13.0	-0.218
Oxygen	154.6	50.4	+0.025

For gas mixtures, the attraction and repulsion between molecules of different components causes non-linear variation of some properties with composition. To account for this in Equation (6), a set of mixing rules can be defined to this non-linearity. The values of "a" and "b" in Equation (6) are re-defined. Based on the recommendations in Reference 9, the following mixing rules are used in the revised PAD code.

$$a_m = \sum_i \sum_j y_i y_j (a_i a_j)^{0.5} (1 - k_{ij}) \tag{11}$$

$$b_m = \sum_i y_i b_i \tag{12}$$

The b_i and a_i for each pure component are given by Equations (7) and (8) respectively. The term k_{ij} is used for some binary pairs to adjust for strong interactions and is determined from experimental data. In the revised PAD code $k_{ij} = 0$, is assumed for all binary combinations.

Attachment 3

Proprietary and Non-proprietary Versions

Of

Response to Question #9 of the NRC's

Request for Additional Information on PAD Model Revisions

Based on the NRC/Westinghouse meeting held on

September 15, 1998

Response to NRC

"Request for Additional Information on PAD Model Revisions"

based on NRC/Westinghouse Meeting

September 15, 1998

Question 9: Please supply typical plots of the following:

a) Corrosion vs. Burnup
b) Clad O.D. vs. Burnup
c) CL-Temperature vs. Power and Burnup
d) Rod Pressure vs. Burnup (IFBA and non-IFBA)
e) Power vs. Burnup

Based on a June 23, 1999 meeting and September 9, 1999 teleconference with NRC and the BNNL reviewer, the requirements and basis of the analysis were further clarified and refined:

Most rods limited by rod internal pressure for Westinghouse cores are IFBA rods; however, the code used by the reviewer for audit calculations does not have IFBA modeling capability. Westinghouse must therefore provide a non-IFBA rod case that generates fuel pressures typical of those seen by rods with IFBA at end-of-life burnups.

1) The typical case is based on a 15x15 lattice non-IFBA fuel rod that generates fuel rod pressures typical of those seen by rods with IFBA (in any Westinghouse fuel lattice) at end-of-life burnups. This case uses peaking factors, power densities, temperatures, flows, chemistry, and power histories, etc., that are representative of existing operating conditions for Westinghouse cores, and is sufficiently aggressive to generate rod pressures that are representative of non-IFBA and IFBA fuel rod duties. The product features of this case were chosen to facilitate the audit calculation by the technical reviewer. Appendix A of this attachment provides the input required for modeling this case in an audit calculation. Information is also provided to allow the reviewers to evaluate the impact of IFBA on fuel rod internal pressure.

2) Westinghouse was asked to provide the specific values for the revised creep model uncertainty, as well as the fuel swelling uncertainty. The creep model uncertainty is addressed in Attachment 4 of the final submittal package. The fuel swelling/densification results are provided below.

3) As agreed upon with the reviewers, all results are based on the creep model described in this submittal package, e.g., with [ ] [a,c], and creep model uncertainties based upon [ ] [a,c] supplied by Westinghouse. Note that a key position of the Westinghouse creep model [ ] [a,c] has been independently reviewed by industry

Figure 1
PAD 4.0 Typical Clad Oxide Thickness Versus Local Burnup

a, c

a, c

Figure 1
PAD 4.0 Typical Clad Oxide Thickness Versus Local Burnup

a, c

Figure 2
PAD 4.0 Typical Clad OD at Temperature Versus Local Burnup

a, c

Figure 3
PAD 4.0 Typical Fuel Centerline Temperature Versus Local Power

a, c

Figure 4
PAD 4.0 Typical Fuel Centerline Temperature Versus Local Burnup

a, c

Figure 5
PAD 4.0 Typical Rod Internal Pressure Versus Rod Average Burnup (MWD/MTU)

a, c

Figure 6
PAD 4.0 Typical Local Power Versus Local Burnup

a, c

Figure 7
PAD 4.0 Typical Pressure Margin Versus Rod Average Burnup

a, c

Figure 8
PAD 3.4 Typical Clad Oxide Thickness Versus Local Burnup

a, c

Figure 9
PAD 3.4 Typical Clad OD at Temperature Versus Local Burnup

a, c

Figure 10
PAD 3.4 Typical Fuel Centerline Temperature Versus Local Power

p

a, c

Figure 11
PAD 3.4 Typical Fuel Centerline Temperature Versus Local Burnup

a, c

Figure 12
PAD 3.4 Typical Rod Internal Pressure Versus Rod Average Burnup

a, c

Figure 13
PAD 3.4 Typical Local Power Versus Local Burnup

a, c

Figure 14
PAD 4.0 Typical Best Estimate and Upper Bound Rod Internal Pressure Difference Between IFBA and non-IFBA versus Rod Average Burnup

a, c

Figure 15
PAD 3.4 Typical Best Estimate and Upper Bound Rod Internal Pressure Difference Between IFBA and non-IFBA versus Rod Average Burnup

a, c

Figure 16
PAD 4.0 Typical Fuel Centerline Temperature Versus Rod Average Burnup at Various Local Powers Used for Power-to-Melt Verification

a, c

Figure 17
PAD 3.4 Typical Fuel Centerline Temperature Versus Rod Average Burnup at Various Local Powers Used for Power-to-Melt Verification

APPENDIX A
TYPICAL FRAPCON-3 INPUT NEEDED FOR A FUEL ROD PERFORMANCE "AUDIT" CALCULATION
RAI-9

The input below is grouped into "rod design" "operating environment" and "operating history". The units shown are convenient for FRAPCON-3 code input.

Table 1: Rod/Assembly Design Parameters

Parameter	Nominal Dimension(s) RAI-9	Units	Comments
Fuel Rod-to-rod pitch	[ ]a, c	inches	Needed to calculate flow channel hydraulic diameter.
Fuel Rod Outer diameter	[ ]a, c	inches	
Cladding wall thickness	[ ]a, c	inches	Alternatively, cladding inner diameter
Cladding inner surface roughness	*	microinches	Used in estimation of minimal thermal gap
Pellet-to-cladding radial gap thickness	[ ]a, c	inches	This is as-fabricated average gap. Need to know reasonable range for this gap based on reasonable combination of dimensional tolerances.
Pellet outer diameter	[ ]a, c	inches	Not needed if gap is specified.
Pellet inner diameter	[ ]a, c	inches	usually zero
Pellet surface roughness	*	microinches	
Pellet dish volume	** [ ]a, c	fraction of total pellet volume	Our code assumes spherical dishes. We would prefer shoulder thickness and dish depth but can deal with just the fractional dish volume.
Pellet chamfer volume	** [ ]a, c	fraction of total pellet volume	Not explicitly modeled, but part of total interface volume.
Pellet as-fabricated density	[ ]a, c	fraction of theoretical density	
Pellet densification or terminal density	[ ]a, c	fraction of theoretical density or fraction of as-fabricated density, or kg per cubic meter.	Please specify units and the test conditions that determined the densification (time, temperature)
Pellet U-235 enrichment	[ ]a, c	Atom % of total U	
Pellet column length	[ ]a, c	inches	
Plenum length	[ ]a, c	inches	
Plenum spring wire diameter	[ ]a, c	inches	Spring geometry is used in plenum gas temperature calculation.
Plenum spring total turns	[ ]a, c	–	
Plenum void volume or total rod internal void volume as-fabricated	[ ]a, c	cubic inches	This is needed as a cross check on the code as-fabricated rod internal volume and volume distribution.
Plenum spring diameter	[ ]a, c	inches	

* [ ]a, c.
** [ ]a, c.

APPENDIX A
TYPICAL FRAPCON-3 INPUT NEEDED FOR A FUEL ROD PERFORMANCE "AUDIT" CALCULATION
RAI-9 (Cont.)

Table 2: Operating Environment Parameters

Parameter	Nominal Dimension(s) RAI-9	Units	Comments
Coolant Inlet Temperature	[ ]^{a,c}	Degrees F	
Coolant Outlet Temperature	[ ]^{a,c}	Degrees F	Used as a cross-check
Coolant mass flow (within the rod's theoretical coolant channel)	[ ]^{a,c}	pound mass per square foot per hour	
Coolant system pressure	[ ]^{a,c}	psia	
Crud thickness or crud deposition rate	[ ]^{a,c}	mils or mils per hour	If no information is available a crud thickness of 0.2 mils will be assumed.
Fast neutron flux level	[ ]^{a,c}	neutrons per square meter per second	If no information is available, we will assume proportionality between specific power (W/gram UO_2) and neutron flux, with the proportionality constant being 0.221 E17 n/m^2/s per W/g.

Table 3: Operating History Parameters

Parameter	Nominal Dimension(s) RAI-9	Units	Comments
Rod-Average end-of-life burnup	[ ]^{a,c}	GWd/MTU	
Axial Peak burnup at EOL	[ ]^{a,c}	GWd/MTU	Used as a cross-check on burnup distribution calculation
Size of each time step (in numbered sequence) or end-of-step cumulative operating time	[ ]^{a,c}	days	
Rod-average linear heat generation rate (LHGR) for each time step	[ ]^{a,c}	kW/ft	
Axial power shape number for each step (which of several shapes do you want us to use for that step?)*	[ ]^{a,c}		
Axial power shape axial station elevations measured from bottom of pellet column	[ ]^{a,c}	feet	
Relative LHGR at each axial station	[ ]^{a,c}	relative power	will be normalized to average value = 1.0

* The axial power shape information must be repeated for each axial power shape used. FRAPCON can use up to 20 power shapes, and up to 20 stations can be used to define each shape. The stations do not need to be equally spaced and the spacing can vary from one shape to the next.

APPENDIX A
TYPICAL FRAPCON-3 INPUT NEEDED FOR A FUEL ROD PERFORMANCE "AUDIT" CALCULATION
RAI-9 (Cont.)

Table 4: Time Dependent Parameters

a, c

APPENDIX A
TYPICAL FRAPCON-3 INPUT NEEDED FOR A FUEL ROD PERFORMANCE "AUDIT" CALCULATION
RAI-9 (Cont.)

Table 5: Axial Power Shapes

a, c

a, c

Figure A-1
Best Estimate Pellet Swelling and Densification (1780 deg C)

Attachment 4

Proprietary and Non-proprietary Versions

Of

Final PAD 4.0 Calibration and V&V Data Package

PAD 4.0

Creep, Thermal, and Fission Gas

Calibration and Verification Statistics

PAD 4.0

Creep Calibration and Verification Statistics

Introduction:

An improved in-reactor irradiation and thermal creep model has been developed[1] and has been incorporated into PAD 4.0. The new creep model is substantially different in form from the model used in PAD 3.4. Therefore, a full calibration, verification and uncertainty analysis was conducted in order to incorporate this new model into PAD 4.0.

The equations that govern the irradiation creep and thermal creep were modified to represent the new formulations. The creep model was developed to accurately model Conventional Zr-4, Improved Zr-4, and ZIRLO™; however, to properly calibrate the model, individual irradiation creep rate and thermal creep rate multipliers were determined for the three alloys from measured profilometry data.

Procedure:

The process used in performing the creep calibration involves comparing the experimentally measured fuel rod profilometry data, with the PAD 4.0 predicted profilometry for each of the calibration rods. Both the measurements and the PAD results include the corrosion layer on the cladding O.D. No attempt has been made to correct the measured data for the oxide layer thickness, as the uncertainties in these corrections would be relatively large. In addition, the PAD cladding O.D. includes the calculated oxide thickness.

The calibration coefficients which determine the thermal and irradiation components of creep are [

]^{a,c}.

Independent calibrations were performed and values of [

]^{a,c}.

Cladding Creep Model Data:

Profilometry data was obtained from [

]^{a,c} and is shown in the table of "List of Creep Data".

Not all of the creep profilometry data can be used for calibration. [

]^{a,c}:

a, b, c

[ ]

Statistical Analysis:

The results of the calibration defined the creep model coefficients for the three alloys are as follows:

Creep Calibration Coefficients

a, b, c

[ ]

The statistical results for the calibration, validation, and verification (total) data for all three alloys are shown below:

Statistical results of creep calibration

a, b, c

[ ]

Figure 1.1 shows the predicted creep-down data versus measured creep-down data for [

]^{a,c}. Figures 1.2 and 1.3 break down the creep data into the calibration and validation subsets.

Figures 1.4 through 1.7 show the residual dependence of the model on axial elevation, rod average burnup, time averaged temperature, and time averaged stress. [

]^{a,c}.

Comparisons of the creep model with individual campaign creep-down data are shown in Figure 1.8 through Figure 1.12 for Conventional Zr-4, Figures 1.13 and 1.14 for Improved Zr-4, and Figure 1.15 for ZIRLO™.

Since the creep model is applicable to both creep-in and creep-out, uncertainties obtained from the verification data will be used in a manner consistent with the creep model. [

]^{a,c}.

Uncertainties:

[

]^{a,c}:

[

]^{a,c}

[

]$^{a, c}$.

The measured, predicted, and weight data are given in tabular form in the List of Creep Data, attached. The weighted standard deviation calculated using the method above was found for both the M/P data and the M-P data. The results are below: [ ]$^{a, b, c}$.

Weighted Standard Deviations of Entire Creep Data Set

[ ]$^{a, b, c}$

[

]$^{a, b, c}$.

The above standard deviations of M/P could be used to calculate the 95% upper bound and 95% lower bound uncertainties in the creep model; however, the scatter in the creep predictions is subject to other uncertainties in addition to the true creep uncertainty. Since some of these other uncertainties can be quantified, they can be removed from the data before using it to calculate upper and lower bounds.

The uncertainty in the predictions can be defined as follows:

[ ]$^{a, c}$

The total uncertainty represented by the standard deviation of the M-P data [ ]$^{a, c}$ is a result of a combination of many uncertainties. The uncertainty of interest is the true creep model uncertainty which will be used to determine the 95% upper bound and 95% lower bound model uncertainties. The measurement uncertainty and as-built fabrication uncertainties are known and given as [ ]$^{a, c}$, respectively. The "other" uncertainties cannot be explicitly defined and can be lumped together with the creep uncertainty. The above equation was solved for the combination of creep and "other" uncertainties giving a value of [

][a,c]. This represents a reduction in the total uncertainty by removing quantifiable uncertainties. The reduction is a factor of [

][a,c]. The uncertainty in the M/P data is what is really needed so it is reduced by this same ratio giving a true creep model M/P uncertainty of [ ][a,c].

The 95% upper bound and 95% lower bound uncertainties were then calculated using a standard deviation of [ ][a,c] along with the individual alloy mean values of M/P. [

][a,c]. The calculations and bounding uncertainties are shown below.

[ ][a, b, c]

Creep Model Uncertainties (ACREEP)

[ ][a, b, c]

Conclusions:

The creep model has been successfully calibrated for use in PAD 4.0. The multipliers for [

][a,c] have been determined and are shown in Creep Calibration Coefficients table. The uncertainties associated with this model are tabulated in preceding table. All of the creep model comparisons show reasonable agreement between measured and predicted data. The results are similar with those determined for the PAD 3.4 creep model.

List of Creep Data

a, b, c

List of Creep Data (cont.)

a, b, c

Figure 1.1

PAD Creep Model Predictions
(all data)

[ ] a, b, c

Figure 1.2

PAD Creep Model Predictions (calibration data)

a, b, c

Figure 1.3

PAD Creep Model Predictions (validation data)

a, b, c

Figure 1.4

PAD Creepdown M-P vs. Axial Elevation (all data)

a, b, c

Figure 1.5

PAD Creepdown M-P vs. Burnup
(all data)

a, b, c

Figure 1.6

PAD Creepdown M-P vs. Time Averaged Temperature (all data)

a, b, c

Figure 1.7

PAD Creepdown M-P vs. Time Averaged Stress (all data)

a, b, c

Figure 1.8

Zion 2,3,4, and 5 cycle Creep Data (Conv. Zr-4)

a, b, c

Figure 1.9

Surry 1,3, and 4 cycle Creep Data (Conv. Zr-4)

a, b, c

Figure 1.10

Trojan 3 cycle Creep Data
(Conv. Zr-4)

a, b, c

Figure 1.11

Farley 1,2, and 4 cycle Creep Data (Conv. Zr-4)

a, b, c

Figure 1.12

North Anna AM2 Creep Data
(Conv. Zr-4)

a, b, c

Figure 1.13

North Anna AM2 Creep Data
(Imp. Zr-4)

a, b, c

Figure 1.14

Vandellos Creep Data
(Imp. Zr-4)

a, b, c

Figure 1.15

North Anna AM2 Creep Data (ZIRLO)

a, b, c

PAD 4.0

Thermal Calibration and Verification Statistics

Introduction:

The thermal model (gap conductance) has not been changed from that licensed in PAD 3.4. However, as a result of changes which were made to revise PAD 4.0 in other models, the fuel rod centerline temperatures required re-calibration. The thermal model calibration was performed in the same manner as was presented in WCAP-8720 licensing submittal for PAD 3.3[2]. Furthermore, the same Halden fuel rod centerline temperature data was used in the calibration as was used in WCAP-8720. The temperature data from Halden assembly IFA-432 was revised from that included in WCAP-8720 using the Halden final data report[3].

The PAD 4.0 gap conductance model, which is the same as the PAD 3.4 gap conductance model for an open gap, is given by:

[ ] a, c

PAD selects the [

] a, c.

Procedure:

The process used in performing the thermal calibration involved comparing the experimentally measured fuel rod centerline temperature data with the PAD 4.0 predicted centerline temperature for each of the calibration rods.

The [

] a, c. This value was used as the starting point for calibration. For each

iteration of the calibration, a new value of [

]$^{a, c}$.

Thermal Model Data:

Fuel temperature data from three instrumented assemblies, IFA-431, IFA-432, and IFA-513, irradiated in the Halden reactor under NRC sponsorship, were used in this calibration. Each assembly contained six fuel rods instrumented with upper and lower thermocouples. Descriptions and pre-characterizations of the fuel are reported in References 4 and 5, and power histories and fuel temperature data are given in References 6, 7, 8, and 9.

The burnup range of the IFA-431, IFA-432, and IFA-513 data used for calibration has been limited. The calibration used thermocouple data in the burnup range of 0 to 5,000 MWD/MTU. There were several reasons for limiting the burnup range. The range of experimental variables in this burnup interval (power, gap size, fuel density, and gas composition) covers the range expected over the entire irradiation of a pressurized PWR fuel rod. Thermocouples de-calibrate as a function of the thermal neutron fluence and there is some uncertainty about the rate of de-calibration. Limiting the burnup range of the investigation also reduces the uncertainties associated with cladding creep rates, fuel densification and swelling rates, and fission gas release rates. Also, most of the high burnup data is from IFA-432 in which considerable fission gas leakage occurred in the rods at higher burnups and thus cannot be precisely modeled. However, the thermal model calibrated from this data is valid for burnups greater than 5,000 MWD/MTU in commercial LWR fuel because the ranges of the fuel variables included in the model derivation cover the range expected at much higher burnups. Furthermore, the validity of the PAD 3.4 (and thus PAD 4.0) thermal model into high burnup values (> 5,000 MWD/MTU) has been addressed in response to NRC questions in WCAP-10851-P-A, PAD 3.4[10].

The measured data contained the following approximate ranges of conditions:

Burnup:	*0 to 5,000 MWD/MTU*
Local Power:	*0 to 14 kW/ft*
Gap:	*0 to 15 mils*
Temperature:	*500 to 3,200 °F*

Data from fifteen rods (three assemblies) from Halden were available for calibration and validation. Eleven rods were selected at random for the temperature calibration. These rods were 431-1, 431-2, 431-5, 431-6, 432-1, 432-3, 432-5, 513-2, 513-3, 513-4, and 513-6. Rods 431-3, 432-2, 513-1, and 513-5 were reserved for validation, and not included in the calibration.

Statistical Analysis:

The PAD code was calibrated using the procedure presented above. The results of the calibration defined the [ ]$^{a, c}$. The statistical results for the calibration, validation, and verification (total) data are shown below.

Statistical Results of Thermal Calibration

[ ] a, b, c

Figures 2.1 and 2.2 show the predicted versus measured thermal data for the calibration and validation subsets respectively. These comparisons show [

] a,c. The results are similar and consistent with those determined for the PAD 3.4 thermal model.

Figures 2.3 through 2.5 show the residual dependence of the model on rod average burnup, local power, and gap size. [

] a,c.

The thermal data has also been evaluated for rods with powers greater than 9kW/ft since these powers are most representative of normal in-reactor operation. Figures 2.6 through 2.9 show the M/P plots for each Halden fuel assembly for data with powers greater than 9kW/ft. Figures 2.10 through 2.12 show the residual dependence of the model on rod average burnup, local power, and gap size. A set of statistics for the greater than 9kW/ft data are given below:

Statistical Results of Thermal Calibration

(Power > 9kW/ft)

[ ] a, b, c

It can easily be seen that the thermal model tends to [

] a,c.

Uncertainties:

The PAD code gives best estimate fuel temperature predictions. The scatter of the differences between measurements and predictions can be due to variations in pre-irradiation physical parameters such as dimensions and densities, model inaccuracies, and instrumentation uncertainties. Since the rods were pre-characterized, it is not expected that a large fraction of the scatter is due to variations of pre-irradiation physical characteristics. However, not all parameters were either pre-characterized or documented and some assumptions were made. In these cases, the assumptions were validated by assuming certain undocumented parameters and comparing the analyzed behavior to the measured fuel. This approach ensured validity of the assumptions. Values of powers for the Halden rods are accurate to ± 3% to ± 5%[11][12] and ± 6%[13] and thermocouple readings are quoted as being accurate to ± 1%[14][15][16].

Skewness and kurtosis tests for normality were conducted on the M-P data and the P-value was determined to be [ ]^{a,c}. The 95% upper-bound and 95% lower-bound uncertainties for the fuel centerline temperatures were calculated to bound 95% [ ]^{a,c} and 5% of the data respectively. The computed uncertainties are shown below.

Thermal Model Uncertainties

[ ]^{a,b,c}

Conclusions:

The thermal model has been successfully calibrated for use in PAD 4.0. The value of [ ]^{a,c}. Associated uncertainties in the thermal model are shown above. All of the thermal model comparisons show reasonable agreement between measured and predicted data. The results for the revised PAD are consistent with those determined for the previously licensed PAD 3.4 thermal model calibration.

Figure 2.1

Measured vs. Predicted Fuel Centerline Temperatures
(calibration data)

a, b, c

Figure 2.2

Measured vs. Predicted Fuel Centerline Temperatures (validation data)

a, b, c

Figure 2.3

Measured - Predicted Fuel Centerline Temperatures vs Burnup (all data)

a, b, c

Figure 2.4

Measured - Predicted Fuel Centerline Temperatures vs Local Power (all data)

a, b, c

Figure 2.5

Measured - Predicted Fuel Centerline Temperatures vs Fuel-to-Cladding Gap (all data)

a, b, c

Figure 2.6

Measured vs. Predicted Fuel Centerline Temperatures
(Assembly 431, power > 9kW/ft)

a, b, c

Figure 2.7

Measured vs. Predicted Fuel Centerline Temperatures (Assembly 432, power > 9kW/ft)

a, b, c

Figure 2.8

Measured vs. Predicted Fuel Centerline Temperatures (Assembly 513-1,2,3, power > 9kW/ft)

a, b, c

Figure 2.9

Measured vs. Predicted Fuel Centerline Temperatures
(Assembly 513-4,5,6, power > 9kW/ft)

a, b, c

Figure 2.10

Measured - Predicted Fuel Centerline Temperatures vs Burnup (power > 9kW/Ft)

a, b, c

Figure 2.11

Measured - Predicted Fuel Centerline Temperatures vs Local Power (power > 9kW/ft)

a, b, c

Figure 2.12

Measured - Predicted Fuel Centerline Temperatures vs Fuel-to-Cladding Gap (power > 9kW/ft)

a, b, c

PAD 4.0

Fission Gas Calibration and Verification Statistics

Introduction:

An improved in-reactor irradiation and thermal creep model has been developed and incorporated into the new PAD 4.0 code. As a result of these and other changes going into the new code, a full calibration, verification and uncertainty analysis was required for both the steady-state and transient fission gas release models.

Procedure:

The intent of this calibration was to compare the PAD 4.0 predicted fission gas release results to measured gas release data. The fission gas calibration was broken into two phases: [

] a, c. The process used in performing the fission gas release calibration required comparing the experimentally measured fission gas release with the PAD 4.0 predicted fission gas release.

[

] a, c.

The steady state fission gas database contains measured data other than just fission gas release. Some of the rods have associated rod growth data, void volume data, and profilometry data. Because the fission gas release model is highly dependent upon having an accurate fuel temperature prediction, it was necessary to be able to model growth, profilometry and void volume of these rods. [

]^{a,c}.

[

]^{a,c}.

[

]^{a,c}.

[

]^{a,c}.

[

]^{a,c}.

Fission Gas Release Model Data:

Fission gas release data from [ ]^{a,c} has been used to verify the steady-state fission gas release model. The range of fabrication and operating conditions covered by these rods is presented below:

[ ] a, b, c

Although the results are not typically included in the statistical evaluation of the steady-state database, measured and predicted release data was reviewed for [ ]$^{a, c}$.

[

]$^{a, c}$.

Rods Eliminated from PAD 3.4

Steady-State FGR Calibration Database

[ ]$^{a, b, c}$

As indicated above, it was decided to [

]$^{a, c}$.

[

]$^{a, c}$.

[

]$^{a, c}$.

[

]^{a,c}.

Statistical Analysis:

The PAD code was calibrated by modifying [

]^{a,c}. The results of the calibration defined the calibration coefficients as follows:

$$\left[\qquad\qquad\qquad \right]^{a,c}$$

A summary of the results of this calibration are presented graphically in Figures 3.4 through 3.15. Figures 3.1, 3.2, and 3.3 show the M/P plots for all of the available fuel rod growth, creep-down, and void volume data associated with the measured fission gas release. [

]^{a,c}. Figures 3.5 and 3.6 compare the predicted and measured fission gas release for the low temperature regime rods. The agreement shown in these figures between the model and the data is reasonable. There is no statistically significant trend of the measured-predicted plots with burnup, showing that the model accounts for the burnup dependence of the fission gas release to the maximum rod average burnup in the data.

[

]^{a,c}.

Figures 3.8 through 3.11 compare the predicted and measured fission gas release for the high temperature steady-state fission gas release data. [

]^{a,c}.

[

]^{a,c}. Comparisons of the measured and predicted fission gas release for these rods are shown in Figures 3.12 through 3.15.

[

] a, c. Furthermore, the results are consistent with what has been predicted for these rods in past evaluations (e.g., PAD 3.4).

The statistical results for the calibration, validation, and verification (total) data are given below. [

] a, c.

Statistical Results of Fission Gas Release Calibration

a, b, c

Uncertainties:

Low Temperature Release Model:

[

] a, c.

[

$]^{a,c}$.

High Temperature Release Model:

Part I: Upper-Bound Uncertainty:

[

$]^{a,c}$.

[

$]^{a,c}$.

[

$]^{a,c}$.

Part II: Lower-Bound Uncertainty:

[

$]^{a,c}$.

Transient Fission Gas Release Uncertainty Analysis:

[

]^{a,c}.

Conclusions:

The steady-state and transient fission gas release models have been successfully calibrated for use in PAD 4.0. All of the fission gas release model comparisons show reasonable agreement between measured and predicted data. The results are similar with those determined for the PAD 3.4 fission gas release model calibration.

A summary of all calibration constants determined in this calibration are given below:

[ ]^{a,b,c}

List of Fission Gas Data

a, b, c

a, b, c

a, b, c

Figure 3.1

Fission Gas Release Database
Fuel Rod Growth Measured vs. Predicted

a, b, c

Figure 3.2

Fission Gas Release Database
Fuel Rod Diametral Creep Down Measured vs. Predicted

a, b, c

Figure 3.3

Fission Gas Release Database
Fuel Rod Void-Volume Measured vs. Predicted

a, b, c

Figure 3.4

All Fission Gas Release Data
Predicted vs. Measured

a, b, c

Figure 3.5

Steady-State Fission Gas Release
Low-Temperature Predicted vs. Measured

a, b, c

Figure 3.6

Steady-State Fission Gas Release
Low-Temperature Measured-Predicted vs. Burnup

a, b, c

Figure 3.7

Fission Gas Release Data (Measured 1-3 %)
Predicted vs. Measured

a, b, c

Figure 3.8

Steady-State Fission Gas Release
High-Temperature Predicted vs. Measured

a, b, c

Figure 3.9

Steady-State Fission Gas Release
High-Temperature Predicted vs. Measured
(measured > 10%)

a, b, c

Figure 3.10

Steady-State Fission Gas Release
High-Temperature Measured-Predicted vs. Burnup

a, b, c

Figure 3.11

Steady-State Fission Gas Release
High-Temperature Measured-Predicted vs. Burnup
(Measured > 10%)

a, b, c

Figure 3.12

Transient Fission Gas Release
Predicted vs. Measured

a, b, c

Figure 3.13

Transient Fission Gas Release Predicted vs. Measured (Measured > 10)

a, b, c

Figure 3.14

Transient Fission Gas Release
Measured-Predicted vs. Burnup

a, b, c

Figure 3.15

Transient Fission Gas Release
Measured-Predicted vs. Burnup
(Measured > 10%)

a, b, c

Figure 3.16

Transient Fission Gas Release: Gadolinia Database Measured-Predicted vs. Burnup (Oconee Rods)

[ ] a, b, c

Figure 3.17

Steady-State Data w/ AFGRL at 2.30
Predicted vs. Measured

a, b, c

Figure 3.18

Steady-State Data w/ AFGRL at 0.47
Predicted vs. Measured

a, b, c

References:

1. "Westinghouse In-Reactor Creep Model," WCAP-15063-P, June 1998.

2. "Revised PAD Code Thermal Safety Model," WCAP-8720, Addenda 2, October 1982.

3. Lanning, D. D., et al., "Irradiation History and Final Post-irradiation Data for IFA-432," NUREG/CR-4717, PNL-5971, November 1986.

4. Hann, C. R., et al., "Test Design, Pre-characterization, and Fuel Assembly Fabrication for Instrumented Fuel Assemblies IFA-431, and IFA-432," NUREG/CR-0332, BNWL-1988, R3, November 1977.

5. Bradley, E. R., et al., "Pre-characterization Report for Instrumented Nuclear Fuel Assembly IFA-513," NUREG/CR-1077, PNL-3156, R-3, November 1979.

6. Hann, C. R., et al., "Data Report for the NRC/PNL Halden Assembly IFA-431," PNL-2494, April 1978.

7. Hann, C. R., et al., "Data Report for the NRC/PNL Halden Assembly IFA-432," NUREG/CR-0560, PNL-2673, 1978.

8. Bradley, E. R., et al., "Data Report for the NRC/PNL Halden Assembly IFA-432: April 1978-May 1980," NUREG/CR-1950, PNL-3709, April 1981.

9. Bradley, E. R., et al., "Data Report for the Instrumented Fuel Assembly IFA-513," NUREG/CR-1838, PNL-3637, August 1981.

10. "Response to NRC Questions on Westinghouse Topical Report, WCAP-10851 (Proprietary), "Improved Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations"," NS-NRC-86-3163, September 30, 1986.

11. Hann, C. R., et al., "Data Report for the NRC/PNL Halden Assembly IFA-431," PNL-2494, April 1978.

12. Bradley, E. R., et al., "Data Report for the Instrumented Fuel Assembly IFA-513," NUREG/CR-1838, PNL-3637, August 1981.

13. Hann, C. R., et al., "Data Report for the NRC/PNL Halden Assembly IFA-432," NUREG/CR-0560, PNL-2673, 1978.

14. Hann, C. R., et al., "Data Report for the NRC/PNL Halden Assembly IFA-431," PNL-2494, April 1978.

15. Hann, C. R., et al., "Data Report for the NRC/PNL Halden Assembly IFA-432," NUREG/CR-0560, PNL-2673, 1978.

16. Bradley, E. R., et al., "Data Report for the Instrumented Fuel Assembly IFA-513," NUREG/CR-1838, PNL-3637, August 1981.

17. WCAP-10851-P-A, "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851-P-A, August 1988.

PAD 4.0

Thermal Modeling of IFA-432 High Burnup Data

PAD 4.0

Thermal Modeling of IFA-432 High Burnup Data

Introduction:

The thermal model calibration of PAD 4.0 was performed in the same manner as was presented in WCAP-8720 licensing submittal for PAD 3.3[1]. The thermal model calibration of PAD 4.0 includes centerline temperature data from Halden IFA-431, IFA-432, and IFA-513. The IFA-432 calibration data includes rods up to 5,000 MWD/MTU. The same IFA-432 rods, used in the PAD 4.0 calibration, were also taken to burnups up to 46,000 MWD/MTU. This assessment documents the ability of PAD 4.0 thermal model to predict the behavior of the high burnup Halden rods.

Measured Fuel Rod Profilometry Data:

Fuel temperature data from instrumented assembly IFA-432 irradiated in the Halden Reactor, under NRC sponsorship, was used in this modeling demonstration. The assembly contained six fuel rods instrumented with upper and lower thermocouples. Descriptions and pre-characterizations of the instrumented fuel are reported in References 2 and 3. Power histories and fuel temperature data for these rods are given in References 4, 5, 6, 7, 8, and 9. IFA-432 rods 1, 2, 3, and 5 were modeled. Rod 6, which contained unstable fuel, and rod 4, which was removed at low burnup, were not modeled. Due to the limited pre-characterization, several fuel rod parameters were assumed based on related Halden experience with other test series. These assumptions were validated against other data provided to ensure that the assumptions were sound. One key assumption was associated with the known fuel rod fission gas leakage that occurred on rods 1, 2, 3, and 5.

The IFA-432 rods were known leakers, presumably through the thermocouple and/or pressure transducer connections. Initially, Westinghouse performed an assessment that modeled the IFA-432 fuel rods assuming that the fission gas leakage occurred. This preliminary assessment, that was provided to the reviewer, showed that the predicted fuel centerline temperatures [ ] a,c. As directed by the NRC/BNNL reviewer, the PAD runs to model this data have assumed that no fission gas leaked from the rods. [

] a,c. This is considered an acceptable approach due to the conservative nature of the "no gas leakage" assumption.

The measured data contained the following approximate ranges of conditions:

Burnup:	0 to 44,000 MWD/MTU
Local Power:	0 to 14 kW/ft
Gap:	-1 to 12 mils
Temperature:	600 to 3,000 °F

Modeling Results:

Table 1:
Statistical results of thermal modeling
Fuel Centerline Temperature Data:

[ ] a, c

Figure 1 shows the predicted versus measured thermal data. This comparison shows [

] a, c

Figures 2 through 4 show the residual dependence of the model on rod average burnup, local power, and gap size. Again, there are [ ] a, c

Figure 5 shows measured versus predicted fuel centerline temperatures as a comparison of the recent IFA-432 data with that of the PAD thermal calibration and validation database. Figures 6, 7, and 8 show the same comparison for the residual dependence of the model on rod average burnup, local power, and gap size.

Conclusions

The thermal model centerline temperature comparisons show a [

] a, c.

Figure 1: IFA-432 Fuel Centerline Temperature Predictions

a, b, c

[ ]

Figure 2: IFA-432 Fuel Centerline Temperature Predictions

a, b, c

Figure 3: IFA-432 Fuel Centerline Temperature Predictions

a, b, c

Figure 4: IFA-432 Fuel Centerline Temperature Predictions

a, b, c

Figure 5: IFA-432 Data Comparison to Calibration Data

a, b, c

Figure 6: IFA-432 Data Comparison to Calibration Data

a, b, c

Figure 7: IFA-432 Data Comparison to Calibration Data

a, b, c

Figure 8: IFA-432 Data Comparison to Calibration Data

a, b, c

References:

1. "Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations," WCAP-8720, October 1976.

2. Hann, C.R., et. Al., "Test Design, Pre-characterization, and Fuel Assembly Fabrication for Instrumented Fuel Assemblies IFA-431, and IFA-432," NUREG/CR-0332, BNWL-1988 R3, November, 1977.

3. Bradley, E. R., et. Al., "Pre-characterization Report for Instrumented Nuclear Fuel Assembly IFA-513," NUREG/CR-1077, PNL-3156, R-3, November, 1979.

4. Hann, C. R., et. Al., "Data Report for the NRC/PNL Halden Assembly IFA-431," PNL-2494, April, 1978.

5. Hann, C. R., et. Al., "Data Report for the NRC/PNL Halden Assembly IFA-432," NUREG/CR-0560, PNL-2673, 1978.

6. Bradley, E. R., et. Al., "Data Report for the NRC/PNL Halden Assembly IFA-432: April 1978-May 1980," NUREG/CR-1950, PNL-3709, April 1981.

7. Bradley, E. R., et. Al., "Data Report for the Instrumented Fuel Assembly IFA-513," NUREG/CR-1838, PNL-3637, August 1981.

8. Lanning, D. D., et. al., "Irradiation History and Interim Post-irradiation Data for IFA-432," NUREG/CR-3071, PNL-4543, March 1984.

9. Lanning, D. D., et. al., "Irradiation History and Final Post-irradiation Data for IFA-432," NUREG/CR-4717, PNL-5971, November 1986.

Attachment 5

Non-proprietary Discussion

Of

Requested SER Language Regarding

"Legacy Fuel" and PAD 4.0 Implementation

As noted in the May 29, 1998 memorandum to Thomas H. Essig (Acting Chief, Generic Issues and Environmental Project Branch, Division of Reactor Program Management, Office of Nuclear Reactor Regulation) from Egan Y. Wang (Reactor System Engineer, Generic Issues and Environmental Project Branch, Division of Reactor Program Management, Office of Nuclear Reactor Regulation), "Meeting Summary of May 5, 1998 Westinghouse Electric Corporation Regarding Fuel Rod Internal Pressure and Other Fuel-Related Activities":

> "The expected results from the improved creep model are more consistent with in-reactor experience using mechanistic approach. Westinghouse stated that for some fuel already in reactor cores, it may be possible that the revised PAD model might still predict some gap reopening. If this were to occur, Westinghouse will demonstrate that these assemblies will continue to meet all safety limits as well as 10 CFR 50.46 oxidation limits for operating as well as future cycles, using the methodology that has already been presented to the NRC for gap reopening analysis. The staff agrees that this is an appropriate way to proceed."

It is requested that the SER reiterate this point with the following additional clarifications:

> "The expected results from the improved PAD 4.0 model are more consistent with in-reactor experience using a mechanistic approach. Westinghouse states that for some fuel already in an operating reactor core or fuel that exists in the spent fuel pool that may be re-inserted in later cycles, it may be possible that the new PAD 4.0 model might still predict some gap re-opening. If analyses were to indicate that this situation could occur, Westinghouse would demonstrate that the affected fuel assemblies will continue to meet all safety limits as well as 10 CFR 50.46 oxidation limits for operating as well as future cycles, using the methodology that has already been presented to the NRC for gap re-opening analysis. The staff agrees that this is an appropriate way to proceed.
>
> Further, it is planned that the implementation of the new PAD 4.0 model will be made on a "forward-fit basis" (e.g., currently analyzed or operating cycles will not require re-analysis using the PAD 4.0 model). All plant specific reload analyses will be analyzed with the new PAD 4.0 model in year 2000 on a schedule consistent with an implementation plan being developed with the Westinghouse Owner's Group. This implementation schedule is based on establishing appropriate documentation and training. The staff finds that this implementation schedule and analysis approach is acceptable"

SECTION M

Westinghouse
Electric Company

Box 355
Pittsburgh Pennsylvania 15230-0355

February 25, 2000
NSBU-NRC-00-5965

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

Attention: J. S. Wermiel, Chief
Reactor Systems Branch
Division of Systems Safety and Analysis

Subject: Westinghouse Improved Performance Analysis and Design Model (PAD 4.0), WCAP-15063-P, Revision 1 (Proprietary), Revisions to RAI #9.

Dear Mr. Wermiel:

Enclosed are copies of the Proprietary and Non-Proprietary versions of the revisions to RAI #9 as a result of discussions held during the review.

Also enclosed are:

1. One (1) copy of the Application for Withholding, AW-00-1389 with Proprietary Information Notice and Copyright Notice.

2. One (1) copy of Affidavit, AW-00-1389.

This submittal contains Westinghouse proprietary information of trade secrets, commercial or financial information which we consider privileged or confidential pursuant to 10 CFR 9.17(a)(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-00-1389 and should be addressed to H. A. Sepp, Manager of Regulatory and Licensing Engineering, Westinghouse Electric Company, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

Henry A. Sepp, Manager
Regulatory and Licensing Engineering

Copy to:
S. L. Wu, NRR
R. Caruso, NRR
S. Bloom, NRR

Westinghouse
Electric Company

Box 355
Pittsburgh Pennsylvania 15230-0355

February 25, 2000
AW-00-1389

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

Attention: J. S. Wermiel, Chief,
Reactor Systems Branch
Division of Systems Safety and Analysis

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Westinghouse Improved Performance Analysis and Design Model (PAD 4.0), WCAP-15063-P, Revision 1 (Proprietary), Revisions to RAI #9.

Reference: Letter from H. A. Sepp to J. S. Wermiel, NSBU-NRC-00-5965, dated February 25, 2000

Dear Mr. Wermiel:

The application for withholding is submitted by Westinghouse Electric Company LLC, a Delaware limited liability company ("Westinghouse"), pursuant to the provisions of paragraph (b)(1) of Section 2.790 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary material for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.790, Affidavit AW-00-1389 accompanies this application for withholding, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

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

Correspondence with respect to this application for withholding or the accompanying affidavit should reference AW-00-1389 and should be addressed to the undersigned.

Very truly yours,

Henry A. Sepp, Manager
Regulatory and Licensing Engineering

Proprietary Information Notice

Transmitted herewith are proprietary and non-proprietary versions of documents furnished to the NRC. In order to conform to the requirements of 10 CFR 2.790 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.790(b)(1).

Copyright Notice

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

Henry A. Sepp

Henry A. Sepp, Manager
Regulatory and Licensing Engineering

Sworn to and subscribed
before me this 25th day
of February, 2000.

Janet A. Schwab

Notary Public

Notarial Seal
Janet A. Schwab, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires May 22, 2000
Member, Pennsylvania Association of Notaries

JANET A. SCHWAB
COMMONWEALTH OF PENNSYLVANIA
NOTARY PUBLIC

(1) I am Manager, Regulatory and Licensing Engineering, in the Nuclear Services Division, of the Westinghouse Electric Company LLC, a Delaware limited liability company ("Westinghouse") and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Electric Company.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0), WCAP-15063-P (Proprietary), Revision to RAI #9," February 25, 2000, for submittal to the Commission. being transmitted by Westinghouse Electric Company (W) letter (NSBU-NRC-00-5965) and Application for Withholding Proprietary Information from Public Disclosure, Henry A. Sepp, Westinghouse, Manager Regulatory and Licensing Engineering to the attention of J. S. Wermiel, Chief, Reactor Systems Branch, Division of Systems Safety and Analysis. The proprietary information is revisions to RAI #9 that resulted from discussion during the review.

This information is part of that which will enable Westinghouse to:

(a) Ensure proper fuel performance of fuel operating in reactors.

(b) Assist customers to obtain license changes resulting from fuel performance modeling.

Further this information has substantial commercial value as follows:

(a) Westinghouse can use this fuel performance modeling capability to further enhance their licensing position over their competitors.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended for developing the enclosed improved core thermal performance methodology.

Further the deponent sayeth not.

Revisions to RAI #9

Proprietary and Non-Proprietary Versions

Response to NRC

"Request for Additional Information on PAD Model Revisions"

based on NRC/Westinghouse Meeting

September 15, 1998

Question 9: Please supply typical plots of the following:

a) Corrosion vs. Burnup
b) Clad O.D. vs. Burnup
c) CL-Temperature vs. Power and Burnup
d) Rod Pressure vs. Burnup (IFBA and non-IFBA)
e) Power vs. Burnup

Based on a June 23, 1999 meeting and September 9, 1999 teleconference with NRC and the BNNL reviewer, the requirements and basis of the analysis were further clarified and refined:

Most rods limited by rod internal pressure for Westinghouse cores are IFBA rods; however, the code used by the reviewer for audit calculations does not have IFBA modeling capability. Westinghouse must therefore provide a non-IFBA rod case that generates fuel pressures typical of those seen by rods with IFBA at end-of-life burnups.

1) The typical case is based on a 15x15 lattice non-IFBA fuel rod that generates fuel rod pressures typical of those seen by rods with IFBA (in any Westinghouse fuel lattice) at end-of-life burnups. This case uses peaking factors, power densities, temperatures, flows, chemistry, and power histories, etc., that are representative of existing operating conditions for Westinghouse cores, and is sufficiently aggressive to generate rod pressures that are representative of non-IFBA and IFBA fuel rod duties. The product features of this case were chosen to facilitate the audit calculation by the technical reviewer. Appendix A of this attachment provides the input required for modeling this case in an audit calculation.

The typical fuel temperature/fuel rod internal pressure case for safety analyses is based on a 17x17 lattice non-IFBA fuel rod that generates fuel rod temperatures and pressures typical of those seen in Westinghouse fuel. This case uses peaking factors, power densities, temperatures, flows, chemistry, and power histories, etc., that are representative of existing operating conditions for Westinghouse cores. The product features of this case were chosen to facilitate the audit calculation by the technical reviewer. Appendix B of this attachment provides the input required for modeling this case in an audit calculation.

2) Westinghouse was asked to provide the specific values for the revised creep model uncertainty, as well as the fuel swelling uncertainty. The creep model uncertainty is addressed in Attachment 4 of the final submittal package. The fuel swelling/densification results are provided below:

3) As agreed upon with the reviewers, all results are based on the creep model described in this submittal package, e.g., with [ ]^{a,c}, and creep model uncertainties based upon [ ]^{a,c} supplied by Westinghouse. Note that a key position of the Westinghouse creep model [ ]^{a,c} has been independently reviewed by industry experts external to Westinghouse. These independent reviews substantiate the Westinghouse position, and are enclosed for your consideration. These reviews are considered proprietary to Westinghouse.

Figures 1 through 7 present PAD 4.0 data for the following parameters for the 15x15 lattice example:

- Typical Clad Oxide Thickness Versus Local Burnup
- Typical Clad OD Temperature Versus Local Burnup
- Typical Fuel Centerline Temperature Versus Local Power
- Typical Fuel Centerline Temperature Versus Local Burnup
- Typical Rod Internal Pressure Versus Rod Average Burnup (Best Estimate and Upper Bound)
- Typical Local Power Versus Local Burnup
- Typical Pressure Margin Versus Rod Average Burnup considering Total, Fuel Densification/Swelling, and Creep Margin effects

Figures 8 through 13 present PAD 3.4 data for the same parameters for comparison purposes. Note that, as expected, the best-estimate fuel rod performance parameters from PAD 4.0 are [ ]^{a,c}, while the upper bound pressures are [ ]^{a,c}. The gap reopening pressure, determined for this case by PAD 4.0, [ ]^{a,c} rod average burnup.

Figures 14 and 15 show the typical best estimate and upper bound rod internal pressure difference between IFBA and non-IFBA fuel rods versus rod average burnup for PAD 4.0 and PAD 3.4, respectively. This data is based on comparisons of 17x17 OFA fuel rods with and without IFBA, and is provided to show the representative rod internal pressure impact of He release due to IFBA on rod internal pressure. The only differences in parameters for this case are uncertainties associated with rod internal pressure based on non-IFBA and IFBA cases - those associated with He release, manufacturing uncertainties on B_{10} loading, and initial backfill pressure [ ]^{a,c}.

Figures 16 and 21 show typical fuel centerline temperature, fuel average temperature and fuel rod internal pressure versus rod average power at various local powers used for power-to-melt verification and safety analyses. As can be

seen by the figures, PAD 4.0 temperature data input to safety analysis is [

]^{a,c}, as expected.

Figure A-1 and B-1 show best estimate pellet sintering and densification effects on pellet density and volume as a function of burnup for each of the audit cases.

Appendix C contains additional clarifications requested to clarify and facilitate the use of the material presented in the audit cases.

a, c

a, c

a, c

a, c

a, c

a, c

APPENDIX A
TYPICAL FRAPCON-3 INPUT NEEDED FOR A FUEL ROD PERFORMANCE "AUDIT" CALCULATION
RAI-9

The input below is grouped into "rod design" "operating environment" and "operating history". The units shown are convenient for FRAPCON-3 code input.

Table 1: Rod/Assembly Design Parameters

Parameter	Nominal Dimension(s) RAI-9	Units	Comments
Fuel Rod-to-rod pitch	[ ]a, c	inches	Needed to calculate flow channel hydraulic diameter.
Fuel Rod Outer diameter	[ ]a, c	inches	
Cladding wall thickness	[ ]a, c	inches	Alternatively, cladding inner diameter
Cladding inner surface roughness	*	microinches	Used in estimation of minimal thermal gap
Pellet-to-cladding radial gap thickness	[ ]a, c	inches	This is as-fabricated average gap. Need to know reasonable range for this gap based on reasonable combination of dimensional tolerances.
Pellet height	[ ]a,c	inches	
Pellet outer diameter	[ ]a, c	inches	Not needed if gap is specified.
Pellet inner diameter	[ ]a, c	inches	usually zero
Pellet surface roughness	*	microinches	
Pellet dish volume	** [ ]a, c	fraction of total pellet volume	Our code assumes spherical dishes. We would prefer shoulder thickness and dish depth but can deal with just the fractional dish volume.
Pellet chamfer volume	** [ ]a,c	fraction of total pellet volume	Not explicitly modeled, but part of total interface volume.
Pellet as-fabricated density	[ ]a, c	fraction of theoretical density	
Pellet densification or terminal density	[ ]a, c	fraction of theoretical density or fraction of as-fabricated density, or kg per cubic meter.	Please specify units and the test conditions that determined the densification (time, temperature)
Pellet U-235 enrichment	[ ]a, c	Atom % of total U	
Pellet column length	[ ]a, c	inches	
Plenum length	[ ]a, c	inches	
Plenum spring wire diameter	[ ]a, c	inches	Spring geometry is used in plenum gas temperature calculation.
Plenum spring total turns	[ ]a, c	–	[ ]a, c
Plenum spring diameter	[ ]a, c	inches	
Total rod internal void volume as-fabricated	[ ]a, c	cubic inches	This is needed as a cross check on the code as-fabricated rod internal volume and volume distribution.
Initial Backfill Pressure	[ ]a, c	psig	[ ]a, c
Axial Mesh – Local Data	6 of 7		

* [ ]a,c.

** [ ]a,c.

APPENDIX A
TYPICAL FRAPCON-3 INPUT NEEDED FOR A FUEL ROD PERFORMANCE "AUDIT" CALCULATION
RAI-9 (Cont.)

Table 2: Operating Environment Parameters

Parameter	Nominal Dimension(s) RAI-9	Units	Comments
Coolant Inlet Temperature	[ ]a, c	Degrees F	
Coolant Outlet Temperature	[ ]a, c	Degrees F	Used as a cross-check
Coolant mass flow (within the rod's theoretical coolant channel)	[ ]a, c	pound mass per square foot per hour	
Coolant system pressure	[ ]a, c	psia	
Crud thickness or crud deposition rate	[ ]a, c	mils or mils per hour	If no information is available a crud thickness of 0.2 mils will be assumed.
Fast neutron flux level	[ ]a, c	neutrons per square meter per second	If no information is available, we will assume proportionality between specific power (W/gram UO_2) and neutron flux, with the proportionality constant being 0.221 E17 n/m^2/s per W/g.

Table 3: Operating History Parameters

Parameter	Nominal Dimension(s) RAI-9	Units	Comments
Rod-Average end-of-life burnup	[ ]a, c	GWd/MTU	
Axial Peak burnup at EOL	[ ]a, c	GWd/MTU	Used as a cross-check on burnup distribution calculation
Size of each time step (in numbered sequence) or end-of-step cumulative operating time	[ ]a, c	days	
Rod-average linear heat generation rate (LHGR) for each time step	[ ]a, c	kW/ft	
Axial power shape number for each step (which of several shapes do you want us to use for that step?)*	[ ]a, c		
Axial power shape axial station elevations measured from bottom of pellet column	[ ]a, c	feet	
Relative LHGR at each axial station	[ ]a, c	relative power	will be normalized to average value = 1.0

* The axial power shape information must be repeated for each axial power shape used. FRAPCON can use up to 20 power shapes, and up to 20 stations can be used to define each shape. The stations do not need to be equally spaced and the spacing can vary from one shape to the next.

APPENDIX A
TYPICAL FRAPCON-3 INPUT NEEDED FOR A FUEL ROD PERFORMANCE "AUDIT" CALCULATION
RAI-9 (Cont.)

a, c

APPENDIX A
TYPICAL FRAPCON-3 INPUT NEEDED FOR A FUEL ROD PERFORMANCE "AUDIT" CALCULATION
RAI-9 (Cont.)

a, c

APPENDIX A
TYPICAL FRAPCON-3 INPUT NEEDED FOR A FUEL ROD PERFORMANCE "AUDIT" CALCULATION
RAI-9 (Cont.)

Table 5: Axial Power Shapes

a, c

Figure A-1
Best Estimate Pellet Swelling and Densification (1780 deg C)

a, c

APPENDIX B
TYPICAL FRAPCON-3 INPUT NEEDED FOR A FUEL ROD PERFORMANCE "AUDIT" CALCULATION
RAI-9 FUELTEMPERATURES/FUEL PRESSURES

The input below is grouped into "rod design" "operating environment" and "operating history". The units shown are convenient for FRAPCON-3 code input.

Table 1: Rod/Assembly Design Parameters

Parameter	Nominal Dimension(s) RAI-9 Case	Units	Comments
Fuel Rod-to-rod pitch	[ ]a,c	inches	Needed to calculate flow channel hydraulic diameter.
Fuel Rod Outer diameter	[ ]a,c	inches	
Cladding wall thickness	[ ]a,c	inches	Alternatively, cladding inner diameter
Cladding inner surface roughness	*	microinches	Used in estimation of minimal thermal gap
Pellet-to-cladding radial gap thickness	[ ]a,c	inches	This is as-fabricated average gap. Need to know reasonable range for this gap based on reasonable combination of dimensional tolerances.
Pellet height	[ ]a,c	inches	
Pellet outer diameter	[ ]a,c	inches	Not needed if gap is specified.
Pellet inner diameter	[ ]a,c	inches	usually zero
Pellet surface roughness	*	microinches	
Pellet dish volume	** [ ]a,c	fraction of total pellet volume	Our code assumes spherical dishes. We would prefer shoulder thickness and dish depth but can deal with just the fractional dish volume.
Pellet chamfer volume	** [ ]a,c	fraction of total pellet volume	Not explicitly modeled, but part of total interface volume.
Pellet as-fabricated density	[ ]a,c	fraction of theoretical density	
Pellet densification or terminal density	[ ]a,c	fraction of theoretical density or fraction of as-fabricated density, or kg per cubic meter.	Please specify units and the test conditions that determined the densification (time, temperature)
Pellet U-235 enrichment	[ ]a,c	Atom % of total U	[ ]a,c
Pellet column length	[ ]a,c	inches	
Plenum length	[ ]a,c	inches	[ ]a,c
Plenum spring wire diameter	[ ]a,c	inches	Spring geometry is used in plenum gas temperature calculation.
Plenum spring total turns	[ ]a,c	--	[ ]a,c
Total rod internal void volume as-fabricated	[ ]a,c	cubic inches	This is needed as a cross check on the code as-fabricated rod internal volume and volume distribution.
Plenum spring diameter	[ ]a,c	inches	

* [ ]a,c.

** [ ]a,c.

APPENDIX B
TYPICAL FRAPCON-3 INPUT NEEDED FOR A FUEL ROD PERFORMANCE "AUDIT" CALCULATION
RAI-9 FUEL TEMPERATURES/FUEL PRESSURES (Cont.)

Table 2: Operating Environment Parameters

Parameter	Nominal Dimension(s) RAI-9 Safety Case	Units	Comments
Coolant Inlet Temperature	[ ]^{a,c}	Degrees F	
Coolant Outlet Temperature	[ ]^{a,c}	Degrees F	Used as a cross-check (5.445 kw/ft)
Coolant mass flow (within the rod's theoretical coolant channel)	[ ]^{a,c}	pound mass per square foot per hour	
Coolant system pressure	[ ]^{a,c}	psia	
Crud thickness or crud deposition rate	[ ]^{a,c}	mils or mils per hour	If no information is available a crud thickness of 0.2 mils will be assumed.
Fast neutron flux level	[ ]^{a,c}	neutrons per square meter per second	If no information is available, we will assume proportionality between specific power (W/gram UO_2) and neutron flux, with the proportionality constant being 0.221 E17 n/m^2/s per W/g.

Table 3: Operating History Parameters

Parameter	Nominal Dimension(s) RAI-9 Safety Case	Units	Comments
Rod-Average end-of-life burnup	[ ]^{a,c}	GWd/MTU	
Axial Peak burnup at EOL	[ ]^{a,c}	GWd/MTU	Used as a cross-check on burnup distribution calculation
Size of each time step (in numbered sequence) or end-of-step cumulative operating time	[ ]^{a,c}	days	
Rod-average linear heat generation rate (LHGR) for each time step	[ ]^{a,c}	kW/ft	
Axial power shape number for each step (which of several shapes do you want us to use for that step?)*	[ ]^{a,c}		
Axial power shape axial station elevations measured from bottom of pellet column	[ ]^{a,c}	feet	
Relative LHGR at each axial station	[ ]^{a,c}	relative power	will be normalized to average value = 1.0

* The axial power shape information must be repeated for each axial power shape used. FRAPCON can use up to 20 power shapes, and up to 20 stations can be used to define each shape. The stations do not need to be equally spaced and the spacing can vary from one shape to the next.

APPENDIX B
TYPICAL FRAPCON-3 INPUT NEEDED FOR A FUEL ROD PERFORMANCE "AUDIT" CALCULATION
RAI-9 FUEL TEMPERATURES/FUEL PRESSURES (Cont.)

[ ]a, c

APPENDIX B
TYPICAL FRAPCON-3 INPUT NEEDED FOR A FUEL ROD PERFORMANCE "AUDIT" CALCULATION
RAI-9 FUEL TEMPERATURES/FUEL PRESSURES (Cont.)

a, c

APPENDIX B
TYPICAL FRAPCON-3 INPUT NEEDED FOR A FUEL ROD PERFORMANCE "AUDIT" CALCULATION
RAI-9 FUEL TEMPERATURES/FUEL PRESSURES (Cont.)

a, c

APPENDIX B
TYPICAL FRAPCON-3 INPUT NEEDED FOR A FUEL ROD PERFORMANCE "AUDIT" CALCULATION
RAI-9 FUEL TEMPERATURES/FUEL PRESSURES (Cont.)

a, c

APPENDIX B
TYPICAL FRAPCON-3 INPUT NEEDED FOR A FUEL ROD PERFORMANCE "AUDIT" CALCULATION
RAI-9 FUEL TEMPERATURES/FUEL PRESSURES (Cont.)

a, c

Figure B-1
Best Estimate Pellet Swelling and Densification (1700 deg C)

a, c

Appendix C

Responses to Requests for RAI-9 Additions and Clarifications

Question 1: **Since we have absolute dish and chamfer parameters rather than pellet column volume fractions, I need the pellet height. It would be nice to have the dish and chamfer volume fraction in addition, just as a cross-check.**

Response 1: Table 1 has been updated to contain the pellet height for both Appendix A and B.

Question 2: **The quoted range for fuel/cladding roughnesses are quite large [ ]^{a,c}. The same range is quoted for both fuel and cladding – is this range the combined roughness values or individual, e.g., is it [ ]^{a,c}? Also, how do you chose for a calculation whether to use [ ]^{a,c}?**

Response 2: The effective fuel/cladding surface roughness is inherent in the thermal model calibration. The annular gap reduction factor is varied until the calibration is achieved, while the effective surface roughness is assumed.

Question 3: **It is not completely clear whether the void volume quoted is plenum-only or total rod internal void volume. I suspect it is the latter.**

Response 3: Appendix A Table 1 has been updated to clarify that the volume supplied is the total rod internal void volume.

Question 4: **It would be good to clarify where BOL and EOL are on the fuel centerline traces (Figures 3 and 10).**

Response 4: Tabular data for Figures 3 and 10 is included for clarification.

Question 5: **Fuel pellet densification continues (at a reduced rate) after 5000 MWd/MTU (Figure A-1). We would have to modify our densification model to track this. I am inclined to simply set our terminal densification at about 1.25%TD for this case.**

Response 5: Westinghouse concurs with the terminal densification assumption at about 1.25% TD for the audit case described in Appendix A.

Question 6: **The power-to-fast flux conversion appears to be very high?? Based on our conversion of units it gives about 1.2×10^{15} n/cm^2/s at 10 kW/ft -- about 10 times more than I would think. Is our interpretation/conversion of your fast flux conversion factor correct?**

Response 6: Appendix A Table 4 has been updated with the appropriate adjustment to correct the fast flux and fluence conversions.

Question 7: **Does your calculation of best estimate rod pressure in Figures 5 and 12 include modeling of transients?**

Response 7: The best estimate rod internal pressure in Figures 5 and 12 consider both Condition I and Condition II transients. The case that is the basis for Figures 1 through 13 includes Condition I transients, Figures 14 through 21 include Condition II Transients.

Question 8: **We are trying to interpret Figure 7. Does it intend to show the reduction in pressure margin to your pressure limit as a result of including the creep and fuel densification/swelling uncertainty factors? Your nominal margin should include best estimate predictions of creep and fuel densification/swelling. Is this interpretation correct?**

Response 8: The nominal margin line, as displayed in Figure 7 is the pressure limit minus the best estimate case run. The fuel densification/swelling margin line represents the pressure limit minus the nominal case plus the fuel densification/swelling uncertainty. The creep margin line represents the pressure limit minus the nominal case plus the creep uncertainty. [

]^{a,c}.

Question 9: **.I have a simple question of clarification about the rod pressure plots that are provided in Figures 5 and 12. Based on the high pressures in these figures they must be typical for an IFBA rod. If so, are these pressures from your sample PAD calculations input provided in Appendix A? I first thought that these pressures were from the PAD audit çalc but on reflection and the high pressures from these figures I am not sure. Also, Figures 14 and 15 indicate that there is a [ ]^{a,c} difference between IFBA and non-IFBA rods. Can you give me a quick answer to clarify the pressures in Figures 5 and 12?**

Response 9: Figures 5 and 12 are from the sample input provided in Appendix A.

Question 10: **This question is in relation to Appendix A of RIA-9, which is input for "fuel rod performance audit calculation: It's a small matter, but can you please verify the number of spring turns in the plenum space spring? Appendix A lists this at [ ]^{a,c}, which is about 4 times less than I would have thought for a compressed ordinary compression spring, about three times less even for an uncompressed spring. It matters a bit to the void volume, hence to the calculated rod internal pressure.**

Response 10: The plenum springs modeled both in Appendix A and B have variable pitches. The cold plenum spring volume has also been provided for both cases.

Question 11: **I looked at the power histories you provided for the example rod pressure calcs in Appendix A and I don't see the transients that you have said (in conference call on 1-3-00) were included in the best estimate PAD predictions in Figures 5 and 12. I need these transients so I can do a direct comparison to the best estimate results in Figures 5 and 12. I would like to reiterate that I need the identical input that was used in PAD 4.0 to calculate the best estimate results you have shown in Figures 1, 2, 4, 5 and 16. I assume that the PAD 3.4 calculations used the same input as was used for PAD 4.0. Is this assumption correct?**

Response 11: Appendix A contains the modeled transients for Figures 1, 2, 4, and 5 and Appendix B contains the transients for Figures 16, 18 and 20. PAD 3.4 inputs replicate the equivalent cases that have been modeled with PAD 4.0.

Question 12: **If the PAD calculation for LOCA input is different I need this PAD input along with the PAD output of centerline and fuel average temperatures and rod pressures provided for the LOCA analyses. I assume that your nominal density for the Appendix B design is about 95.5%TD is this true? I suspect that density will be the largest contributor to the [ ]^{a,c} that you say is the conservatism in the base deck calc for LOCA. This could amount to [ ]^{a,c}. This will help me justify your PAD 4.0 LOCA predictions. Also, even though you seem to be conservative compared to FRAPCON-3 at BOL (<1000 GWD/MTU) I have noticed something interesting in your comparison to the thermal data. From examination of Figure 2.10 of Attachment 4 (at heat ratings > 9kw/ft) [**

]^{a,c}. I am somewhat confident in the FRAPCON-3 predictions at BOL because we have compared them to several more Halden rods (within other instrumented fuel assemblies, IFAs) with thermal couples at BOL than just the 5 rods and two IFAs we presented in our Vol. 3 assessment document of FRAPCON-3. The standard deviation on FRAPCON-3 centerline predictions to this data at BOL is only about 30 °C. However, I always believe experimental data over calculations even if it is with my own code. I am writing the section on your thermal predictions to data.

Response 12: The difference in fuel average temperatures range from [

]^{a,c}.

Errata for Attachment 4

Of

NSBU-NRC-99-5956 Submittal to NRC

Proprietary and Non-Proprietary Versions

ERRATA FOR Attachment 4
Of
NSBU-NRC-99-5956

PAD 4.0
Creep, Thermal, and Fission Gas
Calibration and Verification Statistics

The following change (revision bars and underlined) should be made to page 5 of 68 of Attachment 4 to NSBU-NRC-99-5956:

The 95% upper bound and 95% lower bound uncertainties were then calculated using a standard deviation of [ ]^{a,c} along with the individual alloy mean values of M/P. [ ]^{a,c}. The calculations and bounding uncertainties are shown below.

The following addition (revision bars and underlined) should be made to page 26 of 68 of Attachment 4 to NSBU-NRC-99-5956:

Skewness and kurtosis tests for normality were conducted on the M-P data and the P-value was determined to be [ ]^{a,c}. The 95% upper-bound and 95% lower-bound uncertainties for the fuel centerline temperatures were calculated to bound 95% [ ]^{a,c} and 5% of the data respectively. <u>The fuel average temperature uncertainties were determined in a consistent manner as they were documented in WCAP-8720, "Improved Analytical Models used in Westinghouse Fuel Rod Design Computations". Therefore, the average temperature uncertainty was calculated as [ ]^{a,c} the fuel centerline temperature uncertainty. It is recognized that temperature predictions used as initial conditions for LOCA model calculations are generally limiting at low burnup and at powers greater than 9 kW/ft. In order to increase thermal uncertainty conservatism for these calculations, the fuel volume average uncertainties were also determined using a multiplier of [ ]^{a,c} the fuel centerline temperature uncertainty. [ ]^{a,c}. All of the uncertainties are tabulated below.</u>

Thermal Model Uncertainties

[ ] a, b, c