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Our ref: LTR-NRC-08-36
August 22, 2008

Subject: Follow-Up Response to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, "POLCA-T: System Analysis Code with Three-Dimensional Core Model" (TAC No. MD5258) (Proprietary/Non-Proprietary)

Enclosed are copies of the Proprietary and Non-Proprietary versions of the Follow-Up Response to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, "POLCA-T: System Analysis Code with Three-Dimensional Core Model" (TAC No. MD5258) (Proprietary/Non-Proprietary)

Also enclosed is:

1. One (1) copy of the Application for Withholding, AW-08-2461 (Non-proprietary) with Proprietary Information Notice.
2. One (1) copy of Affidavit (Non-proprietary).

This submittal contains proprietary information of Westinghouse Electric Company, LLC. In conformance with the requirements of 10 CFR Section 2.390, as amended, of the Commission's regulations, we are enclosing with this submittal an Application for Withholding from Public Disclosure and an affidavit. The affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

Correspondence with respect to the affidavit or Application for Withholding should reference AW-08-2461 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. A. Gresham

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Enclosures
cc: A. Mendiola, NRR
P. Yarsky, NRR
G. Bacuta, NRR

T007
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Our ref: AW-08-2461
August 22, 2008

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-NRC-08-36 P-Enclosure, Follow-Up Response to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, "POLCA-T: System Analysis Code with Three-Dimensional Core Model" (TAC No. MD5258) (Proprietary)

Reference: Letter from J. A. Gresham to Document Control Desk, LTR-NRC-08-36, dated August 22, 2008

The application for withholding is submitted by Westinghouse Electric Company LLC (Westinghouse) pursuant to the provisions of paragraph (b)(1) of Section 2.390 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 2.390, Affidavit AW-08-2461 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 2.390 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference AW-08-2461 and should be addressed to J. A. Gresham, Manager of Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Cc: A. Mendiola, NRR
G. Bacuta, NRR

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared J. A. Gresham, 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 (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:

J. A. Gresham

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 22 day
of August, 2008.

Margaret L. Gonano
Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Margaret L. Gonano, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Jan. 3, 2010
Member, Pennsylvania Association of Notaries

(1) I am Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, Westinghouse Electric Company LLC (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 Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse "Application for Withholding" accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse 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 2.390 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 2.390, 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 LTR-NRC-08-36 P-Enclosure, Follow-Up Response to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, "POLCA-T: System Analysis Code with Three-Dimensional Core Model" (TAC No. MD5258) (Proprietary), for submittal to the Commission, being transmitted by Westinghouse letter (LTR-NRC-08-36) and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse Electric Company is responses to NRC's Request for Additional Information.

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

(a) Obtain generic NRC licensed approval for use of the advanced dynamic system analysis code POLCA-T in performing BWR licensing analysis.

(b) Specific applications using the POLCA-T computer code will include Control Rod Drop Accident (CRDA) analysis and BWR stability analysis

Further this information has substantial commercial value as follows:

(a) Future applications of the POLCA-T computer code will include BWR Transient Analysis and Anticipated Transient Without Scram (ATWS) analysis.

(b) Assist customers to obtain license changes.

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 fuel design 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.

## PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 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 2.390(b)(1).

## COPYRIGHT NOTICE

# LTR-NRC-08-36 NP-Enclosure, Follow-Up Response to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, "POLCA-T: System Analysis Code with Three-Dimensional Core Model" (TAC No. MD5258) (Non-Proprietary)

Westinghouse Electric Company
P.O. Box 355
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NRC RAI 6-5

Figure B.7-2 shows uncertainty bands that are reportedly one standard deviation. These bands do not appear to be [[ ]] away from the 45-degree line. Clarify what the bands represent and bring the figure and text into agreement in the next revision to the topical report.

Westinghouse Response to NRC RAI 6-5

The one sigma lines are constructed as [ ]$^{a,c}$ . That means that for each value along the [ ]$^{a,c}$ is constructed by independently applying two uncertainties, [ ]$^{a,c}$. The band is then formed as [ ]$^{a,c}$ . This is a standard technique to be applied when analyzing data with independent uncertainties in measured and calculated values, as can be seen in 'Probability and Statistics in Particle Physics' by A.G. Frodesen, O. Skjeggestad and H. Tøfte (ISBN 82-00-01906-3).

With respect to the additional question posed during the POLCA-T NRC Audit the parameters of [ ]$^{a,c}$, which are used to estimate the decay ratio from noise measurements, are calculated by minimizing the following loss function:

[ ]$^{a,c}$

where:

[ ]$^{a,c}$.

The covariance matrix of θ is then calculated, and using the Gauss approximation formula, the decay ratio variance is estimated from the uncertainty of the [ ]$^{a,c}$. The representative uncertainty used in WCAP-16747-P is calculated from [ ]$^{a,c}$.

The one sigma decay ratio uncertainty is then verified with the OECD-benchmark, shown in Figure B.7-1 in the Topical Report.

RAI 6-17

Separately determine the uncertainty for each oscillation mode and justify the use of a single value for the acceptance criterion. The staff notes that many more core wide or global oscillations were considered than channel or regional oscillations.

Westinghouse Response to NRC RAI 6-17

In standard stability analysis such as backup stability protection, the decay ratio is used as the stability measure. That is, the relation between consecutive maxima in an impulse response is estimated. Stability regions for decay ratios below 1.0 are searched, for both the global and the regional modes. That is, the behavior of the core is expected to be linear in amplitudes for both modes of oscillations. Since the database covers global decay ratios up to the stability limit (1.0) and a regional case at the stability limit using the same fundamental physics, the code can handle both modes [ ]$^{a,c}$ can be used. Above the stability limit, when amplitudes may become large, detect and suppress systems are used for stability protection. The detect and suppress system relies on the correlation between oscillation amplitude and dry out (DIVOM), which is a different phenomenon [ ]$^{a,c}$.

NRC RAI 6-35

In response to RAI 6-12 the fraction of nominal flow rate was provided for KKL, however, the staff requested that the absolute flow rate also be provided. See NRC RAI 6-12. Please supplement the response to RAI 6-12 with the requested information in cases where it is available.

Westinghouse Response to NRC RAI 6-35

The nominal flow in KKL is [ ]$^{a,b,c}$ kg/s. This is an absolute value, the definition has not been changed in connection with power up-rates etc., and is therefore applicable for all measurements.

RAI 7: Control Rod Drop Accident (CRDA)

The staff has several questions regarding Appendix A of the topical report, which discusses the application of POLCA-T to CRDA analysis.

NRC RAI 7-4

The sensitivity study in A.5.1.3 concludes that the peak fuel enthalpy is insensitive to the delayed neutron fraction within 20%. The LTR states that this is consistent with the previously approved method (RAMONA-3B SCP2). Please reconcile the statement in the subject LTR with the figure produced in A.3-1 of CENPD-284-P-A. Refer to BNL-NUREG-66230 and BNL-NUREG-67430, describe those aspects of the POLCA-T methodology that result in an insensitivity to delayed neutron fraction while previous sensitivity studies indicate a large sensitivity to delayed neutron fraction.

Westinghouse Response to NRC RAI 7-4

In order to perform the sensitivity study a delayed neutron fraction multiplier has been implemented. The incorrect implementation of this multiplier resulted in POLCA-T insensitivity to the delayed neutron fraction. Delayed neutron fractions have been multiplied only once at the initial steady-state; however, the POLCA-T code updates the kinetics data at each time step to account for state parameters change (including control rod position). Thus when used in the transient simulations, delayed neutron fractions have not been multiplied and the code was insensitive to them. The issue has been resolved.

Hereafter, the results obtained with the correctly implemented delayed neutron fraction multiplier will be provided and discussed with corresponding corrections in Appendix A of the Topical report.

The text of section A.5.1.3 on pages A-60 to A-61 (including Table A.5-2) will be replaced by the following:

*A.5.1.3 Delayed Neutron Fraction*

*POLCA-T treats the effective delayed neutron fractions on a nodal basis similar to the way of treatment of the cross-section data. [*

*]$^{a,c}$ Linear Least Square Fit provided in Figure A.5-2a will be later used to derive the POLCA-T peak fuel enthalpy uncertainty due to delayed neutron fraction variations.*

a,c

a,c

Figure A.5-2a. POLCA-T Predicted Peak Fuel Enthalpy and Linear Least Square Fit versus Delayed Neutron Fraction Multiplier

*Figures A.5-2b through A.5-2e present the POLCA-T predicted fission power, total reactivity, peak fuel enthalpy and maximum hot rod fuel centerline temperature time histories at different delayed neutron fraction multipliers. [*

]$^{a,c}$

a,c

[

]

Figure A.5-2b. POLCA-T Predicted Fission Power Time Histories at Different Delayed Neutron Fraction Multipliers

a,c

Figure A.5-2c. POLCA-T Predicted Total Reactivity Time Histories at Different Delayed Neutron Fraction Multipliers

a,c

Figure A.5-2d. POLCA-T Predicted Peak Fuel Enthalpy Time Histories at Different Delayed Neutron Fraction Multipliers

a,c

Figure A.5-2e. POLCA-T Predicted Maximum Hot Fuel Rod Centerline Temperature Time Histories at Different Delayed Neutron Fraction Multipliers

*Figure A.5-2f provides the comparison of POLCA-T and RAMONA-3B SCP2 peak fuel enthalpy sensitivities to delayed neutron fraction variations. The figure produced in A.3-1 of CENPD-284-P-A (page 145) has been used for RAMONA-3B SCP2 sensitivity.* [

]$^{a,c}$.

[ ]a,c

Figure A.5-2f. POLCA-T and RAMONA3B Peak Fuel Enthalpy sensitivity to Delayed Neutron Fraction[1]

*Figure A.5-2g shows the POLCA-T peak fuel enthalpy sensitivity to delayed neutron fraction versus control rod worth. This comparison is similar to those provided in Figures 2 of BNL-NUREG-66230 (Reference 1) and BNL-NUREG-67430 (Reference 2). The data points on the graph represent POLCA-T results for energy deposited during the power pulse. POLCA-T results provided in Figure A.5-2g demonstrate [*

*]a,c.*

*The curve in the figure is calculated using a simplified expression for the sensitivity S=-1/(R-1) derived from the point kinetics model and Nordheim-Fuchs approximation. It is also shown in References 1 and 2. The figure shows that the results for the power pulse are [*

*]a,c.*

---

[1] [

]a,c

a,c

Figure A.5-2g. POLCA-T Peak Fuel Enthalpy sensitivity to Delayed Neutron Fraction versus Control Rod Worth

Item 7 in section A.5.1.12, "Summary on the Sensitivities Studies," on page A-70 will be replaced by following:

a,c

Section A.5.3.3 on page A-74 will be replaced by following

*A.5.3.3 [* *]$^{a,c}$*

*[*

*]$^{a,c}$*

[

]a,c.

Section A.5.3.7 [ ]a,c on page A-75 and Table A.5-10 on page A-76 will also be corrected to include the above uncertainty of peak enthalpy due to the delayed neutron fraction.

**References:**

[1]. D. J. Diamond, Ch-Y. Yang, A. L. Aronson, Estimating the Uncertainty in Reactivity Accident Neutronic Calculations, BNL-NUREG-66230.

[2]. D. J. Diamond, A. L. Aronson, Ch-Y. Yang, A Qualitative Approach to Uncertainty Analysis for the PWR Rod Ejection Accident, BNL-NUREG-67430.

[3]. Control Rod Drop Accident Analysis for Boiling Water Reactors: Summary and Qualification, ABB CE Topical Report CENPD-284-P-A (Proprietary), July 1996.

NRC RAI 7-9

Describe any controls on the time step or other controls in the iterative solution technique that ensure sufficient nuclear power distribution iterations between thermal hydraulic iterations to ensure that the transient pin power distribution is adequately characterized to determine the integrated hot pin energy deposition during CRDAs.

Westinghouse Response to NRC RAI 7-9

Details about POLCA-T time size control algorithm are provided in the answer to RAI 4-8. The time step size control approach that is specific for reactivity transients is discussed in the answer to RAI 4-9. It was mentioned that a sensitivity study is performed in order to determine the suitable upper limitation of the time step which is normally set to a few milliseconds. Example analyses indicating that the time step control does not adversely impact the numerical results of the transient reactor behavior predicted by POLCA-T were provided in the answer to RAI 6-3 (see also Reference 2).

The above mentioned information provides a background to the RAI response provided hereafter.

- Kinetics - thermal-hydraulics iteration scheme

The 3-D-kinetics model is solved using an iterative method and is then iterated in an outer loop including the thermal-hydraulic equations until convergence is reached during transients. If the solution does not converge, [ ]$^{a,c}$ is made until convergence is reached. If no convergence is achieved and [ ]$^{a,c}$ is at its lowest allowed size the code stops, with an alarm message. This procedure leads to a consistent solution of power generation and thermal-hydraulics, i.e. consistent solution between power and reactivity feedback. Thereafter the enthalpy can be evaluated by [ ]$^{a,c}$ for each fuel rod at different axial elevations. Figure 1 shows the outline of the computational procedure in POLCA-T.

- POLCA-T CRDA sensitivity to time step size

In order to investigate the sensitivity to time step size of POLCA-T results on nodal and pin level during CRDA simulation a set of calculations had been performed with increasing time step upper limit. The simulation with [ ]$^{a,c}$ provides a reference case with constant time step, i.e. no effect of the time step size algorithm on the results is assumed in this case. The other time step upper limits selected were [ ]$^{a,c}$. A summary of the obtained results is presented in Table 1.

a,c

[ ]

**Figure 1. Time integration of the power - thermal hydraulic interaction**

a,c

[ ]

As show in Table 1 the variation of the time step upper limit between [ ]$^{a,c}$ only slightly affects the timing of [ ]$^{a,c}$. Their variation is limited to [ ]$^{a,c}$. While the peak power value changed by roughly [ ]$^{a,c}$, the value of the maximum fuel enthalpy variation is limited to [ ]$^{a,c}$.

The deviations of the values of peak power, maximum fuel enthalpy, maximum hot rod node average and centerline fuel temperatures to the ones observed in the reference case with constant time step of [ ]$^{a,c}$ are provided in Table 2. It is observed that while the predicted [ ]$^{a,c}$ is affected by the time step variation, the [ ]$^{a,c}$ only show insignificant variations.

a,c

[ ]

Figures 2 through 4, show POLCA-T predicted time histories for fission power, hot rod peak fuel enthalpy and the maximum hot rod fuel centerline temperature at different time step upper limits. [

]a,c

a,c

[ ]

Figure 2. POLCA-T predicted Fission Power Time Histories át different time step upper limits

a,c

[ ]

Figure 3. POLCA-T predicted Peak Fuel Enthalpy Time Histories at different time step upper limits

a,c

[ ]

Figure 4. POLCA-T predicted Maximum Hot Fuel Rod Centerline Temperature Time Histories at different time step upper limits

Figure 5 illustrates the actual time step produced by the POLCA-T time step algorithm at different time step upper limits. [

][a,c].

As it was mentioned in the answer to RAI 4-9, the POLCA-T CRDA methodology states that the code user should specify the time step upper limit that leads to an almost [ ][a,c] to make it sure that the neutron kinetics converges. Due to the very high neutron flux changes that take place in the RIA the upper limitation of the time step is set normally to [ ][a,c] For a given set of analysis the user is also required to investigate the time step effect on the peak power value, [ ][a,c] The time step limit obtained in such a sensitivity study is used in further analyses. Thus it is assured also that the transient pin power distribution is adequately characterized to determine the integrated hot pin energy deposition during CRDA.

a,c

Figure 5. POLCA-T actual time step produced at different time step upper limits

**References**

[1] Ulf Bredolt, " On the Time Integration Method and its Impact on Prediction of Hydraulic Stability by the POLCA-T Code" 15th International Conference on Nuclear Engineering ICONE15, Nagoya, Japan, April 22-26, 2007, paper ICONE15-10033.

## NRC RAI 7-14

Does the POLCA-T methodology account for changes in the pellet dimensions when determining the reactivity worth of Doppler feedback? Specifically, are radial and axial thermal expansion considered? If not, estimate the uncertainty associated with fuel pellet expansion on the predicted peak fuel enthalpy. This estimate may be based on an analysis using PHOENIX to determine a bias in the Doppler coefficient. Additionally, when evaluating the negative Doppler feedback, does POLCA account for increased resonance absorption in all nuclides? If so, are there any volatile nuclides in the fuel that contribute significantly to the negative reactivity feedback? If so are the release mechanisms for these volatile nuclides considered?

## Westinghouse Response to NRC RAI 7-14

The POLCA-T methodology [ ]$^{a,c}$ when determining the fuel temperature reactivity feedback. This is mainly because the currently implemented models [ ]$^{a,c}$ changes in the core dimensions. In particular, this [ ]$^{a,c}$ that the modeling of axial fuel expansion must be consistent with the used core dimensions in the 2-dimensional calculations of cross sections for POLCA-T. Since the principal output of 2D calculations are macroscopic cross sections, applying the [ ]$^{a,c}$ of fuel in 2D models with the correspondingly reduced [ ]$^{a,c}$ while neglecting the [ ]$^{a,c}$ changes of core [ ]$^{a,c}$ in 3D calculations, would account for the part of the [ ]$^{a,c}$ that goes outside the [ ]$^{a,c}$ of the 3D model. The radial expansion of pellets could be added to the calculations using current models, [ ]$^{a,c}$ due to its rather small effect (cf. Reference [1]) which does not justify the complications that would have to be introduced in the computational scheme for [ ]$^{a,c}$.

For RIA transients that are essentially [ ]$^{a,c}$, the uncertainty in peak fuel temperature and enthalpy associated with [ ]$^{a,c}$ can be estimated applying the simplified Nordheim-Fuchs model (References [2, 3, 4]). As an example, Table 1 below shows the calculations of the peak fuel temperature rise after a step reactivity insertion of [ ]$^{a,c}$ performed for a BWR assembly at [ ]$^{a,c}$ void and operational conditions [ ]$^{a,c}$. In this example, the largest error caused by [ neglecting thermal expansion (both radial and axial) ]$^{a,c}$ amounts to approximately [ ]$^{a,c}$ when fuel is depleted to about [ ]$^{a,c}$. It can be seen that the current practice of [ ]$^{a,c}$ is conservative for fresh fuel.

Table 2 shows the maximum peak enthalpy rise for step reactivity insertion in the range of [ ]$^{a,c}$. Maximum temperature errors due to [ ]$^{a,c}$ at different inserted reactivity are obtained by repeating the calculations provided in Table 1. Then the corresponding maximum peak enthalpy rise has been calculated. The maximum peak enthalpy rise both absolute and relative [ ]$^{a,c}$ of inserted reactivity however the ratio of enthalpy rise to inserted reactivity is [ ]$^{a,c}$. This has been used to derive a following least square fit for peak enthalpy correction $\Delta H$ in percentage

[ ]$^{a,c}$

Where $R$ is the reactivity inserted in \$.

a,c

The peak enthalpy correction $\Delta H$ obtained from equation (1) will be used to correct the POLCA-T calculated peak enthalpy. It will be included in the code uncertainty evaluation and at the end of section A.5.3.2 on page A-74 the following text will be added.

[

]a,c

Section A.5.3.7 [ ]$^{a,c}$ on page A-75 and Table A.5-10 on page A-76 will also be corrected to include this uncertainty in peak enthalpy due to the Doppler coefficient sensitivity to thermal expansion.

As mentioned in the answer to RAI 7-1, the PHOENIX4 cross section library contains temperature dependent nuclear data for all important isotopes. Temperature dependent data is available for both basic cross sections, including scattering matrices, and resonance integrals. [ ]$^{a,c}$. In particular, the library has temperature dependent scattering data for oxygen. The temperature range of the tabulations varies, but for the isotopes that may occur in fuel it is from [ ]$^{a,c}$. These extended tabulations guarantee proper treatment of Doppler and other effects in the calculation of fuel temperature feedback.

Concerning the calculation of Doppler coefficients, the PHOENIX4/POLCA-T methodology does [ ]$^{a,c}$ special treatment of [ ]$^{a,c}$ in the fuel. The basic assumption is that [ ]$^{a,c}$. The background of this is twofold:

- in modern BWR fuel, less than [ ]$^{a,c}$ diffuse out of fuel in the practical burnup range, up to about [ ]$^{a,c}$ (Reference [5])

- volatile fission products that are long-lived enough to significantly diffuse out of fuel generally have too [ ]$^{a,c}$ to significantly influence the calculations. Additionally, in transient calculations, the time scale is [ ]$^{a,c}$ any noticeable diffusion.

**References**

[1] J. M. Kallefelz, L. A. Belblidia, P. Grimm, "Fuel-Temperature Coefficient for Boiling Water Reactor Transient Calculations: Its Dependence on Reactor Conditions", Nucl. Sci. Eng., vol. 121, 301-311 (1995)

[2] D. L. Hetrick, "Dynamics of Nuclear Reactors", The University of Chicago Press (1971)

[3] K. O. Ott, R. J. Neuhold, "Introductory Nuclear reactor Dynamics", American Nuclear Society (1985)

[4] D. J. Diamond, A. L. Aronson, Ch-Y. Yang, "A Qualitative Approach to Uncertainty Analysis for the PWR Rod Ejection Accident", BNL-NUREG-67430 (2000)

[5] W. R. Harris et al., "Fuel Rod Design Methods for Boiling Water Reactors – Supplement 1, Volume II", WCAP-15836-P-A Rev. 0 (2006)

## NRC RAI 8-1

*Verify that the critical power correlations included in the POLCA-T dryout correlation library are based on experimental data and not simulated results. Verify that the uncertainties in these correlations are determined from experimental data. The staff will not accept the use of critical power correlations that are not based on experimental data collected from an appropriate full scale test facility. If correlations that are not approved by the NRC exist in the dryout correlation library what controls exist to ensure these correlations are not used in licensing calculations? Please provide a table which contains (1) the dryout correlations in the library, (2) the fuel design that the correlation is applicable to, (3) whether this correlation has been reviewed and approved by the NRC, and (4) the source of the experimental data used to determine the correlation.*

## Westinghouse Response to NRC RAI 8-1

CPR correlations for Westinghouse BWR fuel used for U.S. Licensing Analyses are based on critical power test data and are reviewed by the NRC. The Critical Power uncertainties for these correlations are determined from the same experimental data used to establish the correlation. CPR data for Westinghouse BWR fuel are obtained in the FRIGG Loop in Sweden which is a full scale facility.

The process for establishing CPR correlations for non-Westinghouse Legacy fuel is described in Section 5.3.2.5 of Reference 8-1. In accordance with Restriction 7 of the NRC Safety evaluation for Reference 1, the value of the conservative adder to the OLMCPR for each application is provided to the NRC.

Internal Westinghouse requirements assure the use of NRC-approved correlations for Licensing Analyses in the U.S. Documentation of Licensing Analyses include identification of any CPR correlations(s) used, refer to the NRC-approved correlation documentation, and explain the manner in which NRC Safety Evaluation restrictions or limitations on the correlation used are accommodated.

Westinghouse BWR fuel designs currently operating in U.S. plants are identified in Table 8-1 along with supporting CPR documentation.

Table 8-1

| Correlation | Fuel Type | Reference |
|---|---|---|
| D4.1.1 | SVEA-96 Optima2 | 8-2 |
| ABBD1.0 | SVEA-96 | 8-3 |
| ABBD2.0 | SVEA-96+ | 8-4 |

Proposed improvements to D4.1.1 in References 8-5 and 8-6 are currently under NRC review. In the future as new or improved correlations receive approval from the NRC they will be utilized by POLCA-T and other licensed codes.

**References**

8-1. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel", July 1996.

8-2. WCAP-16081-P-A, "10x10 SVEA Fuel Critical Power Experiments and CPR Correlation: SVEA-96 Optima2", March 2005

8-3. CENPD-392-P-A, "10x10 SVEA Fuel Critical Power Experiments and CPR Correlation: SVEA-96", September 2000.

8-4. CENPD-389-P-A, "10x10 SVEA Fuel Critical Power Experiments and CPR Correlation: SVEA-96+", June 1998.

8-5. WCAP-16081-P-A, Addendum 1: "SVEA-96 Optima2 CPR Correlation (D4):High and Low Flow Applications", November 2006

8-6. WCAP-16081-P-A, Addendum 2: "SVEA-96 Optima2 CPR Correlation (D4):Modified R-factors for Part Length Rods", November 2006

NRC RAI 8-5

Please compare the DF01 and DF02 void quality correlations to the AA78 void quality correlation, please compare the extent of the database in terms of void fraction, pressure, and mass flux used in the development of each correlation. Please refer to WCAP-16606-P. Using the same SVEA-96 test data quantify the uncertainty in the DF01 and DF02 void quality correlations and provide tables substantially similar to Table 3-3. Please also comment on the expected range of pressures that these correlations are applicable to. Justify the future application of the DF01 and DF02 void quality correlations to void fractions above 90% and to pressures above 9MPa.

Westinghouse Response to NRC RAI 8-5

In the DF01 and DF02 correlations the relationship between the phases is given by a constitutive model called drift flux equation. This drift flux equation for the relative motion between the two phases replace the momentum equation for the gas phase, together with a momentum equation for the total mass flow.

The DF01 and DF02 correlations will be compared to AA78 and EPRI void quality correlations. The AA78 slip correlation is described in the BISON Topical Report, Reference 1. This correlation is basically a bubble flow correlation modified to cover annular flow as well as in BWR fuel bundles. It is a best fit to void from the full scale void measurements [ ]$^{a,c}$ performed at the Westinghouse FRIGG test loop, and it was verified to also predict the results from the measurements performed for [ ]$^{a,c}$ Both the DF01 and the DF02 correlations were optimized to fit the [ ]$^{a,c}$ measurements, and validated against the [ ]$^{a,c}$

The parameter ranges in the FRIGG void measurements were:

Table 1. Covered Ranges in the DF01 and DF02 Data Base

[ ]$^{a,c}$

The error distributions and standard deviations for the AA78 slip correlation, DF01 and DF02 void correlations for different void ranges are shown in Table 2 and the comparison against each measurement series in Table 3.

Table 2. Error Distribution as a function of predicted void

a,c

[ ]

a,c

[ ]

a,c

[ ]

Table 3. Mean error and standard deviation of predicted void compared to the measured void for the different Series

a,b,c

| Mean Error and Standard Deviation of the AA78 Void predictions for the Different Series | | | | | |
|---|---|---|---|---|---|
| | | | | | |
| | | | | | |
| | | | | | |
| | | | | | |
| | | | | | |
| | | | | | |

a,b,c

| Mean Error and Standard Deviation of the DF01 Void predictions for the Different Series | | | | | |
|---|---|---|---|---|---|
| | | | | | |
| | | | | | |
| | | | | | |
| | | | | | |
| | | | | | |
| | | | | | |

a,b,c

| Mean Error and Standard Deviation of the DF02 Void predictions for the Different Series | | | | | |
|---|---|---|---|---|---|
| | | | | | |
| | | | | | |
| | | | | | |
| | | | | | |
| | | | | | |
| | | | | | |

As stated in Reference 2, the verified range for pressure and void may be exceeded in some extreme cases. The extrapolation beyond the test conditions is justified by the following facts:

8. [

$]^{a,c}$

9. [

$]^{a,c}$

10. [

$]^{a,c}$

The verified data range covers most BWR applications. However, in some extreme cases, such as design basis pressurization transients (MSIV closure without position scram) or trip of all recirculation pumps, the limits of the above data range may be exceeded. However, the dependencies in pressure and mass flux [ $]^{a,c}$ and the correlation prediction [ $]^{a,c}$

The DF01 and DF02 correlations are, as shown above, verified against measured data for pressures up to [ $]^{a,c}$. The RMS error of the DF01 and DF02 correlations as implemented in POLCA-T are [ $]^{a,c}$ respectively by direct comparisons to measurement data. The corresponding mean errors are [ $]^{a,c}$ respectively. When extrapolating further, [ $]^{a,c}$ a comparison with the EPRI correlation is used.

The EPRI void correlation (equivalent to the Chexal-Lellouche drift flux correlation) is based on a larger data base which includes not only rod bundle measurements but also measurements from heated rectangular channels and round tubes. The description of the correlation is given in Reference 1 and 3. In Reference 3, void fraction results were compared to a wide range of experimental data with various geometry, inlet subcooling, power distribution, and pressure values [ $]^{a,c}$, including measurement in rectangular channel experiments at [ $]^{a,c}$.

Comparisons of void difference to results [ $]^{a,c}$ for different pressures at otherwise constant parameters are shown in Figures 10 - 12 for the EPRI, DF01 and DF02 correlations. The comparison of these figures indicates that the change of void fraction with pressure over the range [ $]^{a,c}$.

The conclusion of the comparison against experimental data demonstrates that there is [ $]^{a,c}$. Therefore there is [ $]^{a,c}$ due to extrapolation of the correlation to pressures higher than those included in the data base [ $]^{a,c}$. This is confirmed by comparing the DF01 and DF02 void to the EPRI void correlation based on experimental data for a wider range of pressures, [ $]^{a,c}$. The comparison between the DF01 and DF02 void correlations to other methods as shown in Figures 10 to 12, indicates that the change in void fraction with pressure predicted over the range [ $]^{a,c}$ at qualities higher than [ $]^{a,c}$.

a,b,c

Figure 1. DF01 prediction errors for all [ ][a,b,c] data points versus pressure

a,b,c

Figure 2. DF02 prediction errors for all [ ][a,b,c] data points versus pressure

a,b,c

Figure 3. DF01 prediction errors for the OF-64B data versus measured void

a,b,c

Figure 4. DF02 prediction errors for the OF-64B data versus measured void

a,b,c

Figure 5. DF01 void as a function of steam quality at [ ]$^{a,c}$

a,b,c

Figure 6. DF02 void as a function of steam quality at [ ]$^{a,c}$

a,b,c

Figure 7. DF01 void as a function of pressure at a mass flux of [ ][a,c] and different steam qualities

a,b,c

Figure 8. DF02 void as a function of pressure at a mass flux of [ ][a,c] and different steam qualities

a,c

Figure 9. Calculated void differences between DF01 and DF02 respectively and the EPRI correlation at [ ]$^{a,c}$ and a steam quality of [ ]$^{a,c}$

a,c

Figure 10. Comparisons of Void Changes as a Function of Steam Quality at Different Pressures for the EPRI void correlation

a,c

Figure 11. Comparisons of Void Changes as a Function of Steam Quality at Different Pressures for DF01

a,c

Figure 12. Comparisons of Void Changes as a Function of Steam Quality at Different Pressures for DF02

## References

[1] G. S. Lellouche, B. A. Zolotar, "A Mechanistic Model for Predicting Two-Phase Void Fraction for Water in Vertical Tubes, Channels, and Rod Bundles, Electric Power Research Institute," EPRI NP-2246-SR, 1982

[2] Westinghouse report WCAP-16606-P-A, "Supplement 2 to BISON Topical Report RPA 90-90-P-A"

[3] Paul Coddington and Rafael Macian, "A Study of the Performance of Void Fraction Correlations used in the Context of Drift-Flux Two-Phase Flow Models," Nuclear Science Engineering and Design, 215 (2002) 199-216

**Request for Additional Information 10: Control Systems**

The staff has questions in regards to the POLCA-T models for control systems.

NRC RAI 10-1

Please provide additional details regarding the modeling of control systems. In particular please describe how POLCA-T models control systems with proportional integral derivative (PID) controllers.

Westinghouse Response to NRC RAI 10-1

There is [ ][a,c] controller (PID) in POLCA-T. However a [ ][a,c] are available instead. In addition there exists a [ ][a,c] for comparing signals starting up a function generator that can control other boundary conditions to the model, such as mass flows. If further functionality is needed, the SAFIR package for control systems can be used.

A [ ][a,c] block with built in function generator are shown below:

a,c

PI-controller:

The input consists of [ ][a,c]. The output can be one signal or several signals.

The [ ][a,c] with built in function generator is schematically shown below:

a,c

The input to this block consists of one or two input signals. The input signal is either compared with the set point value or the two signals are compared to each other.

The operator can be “less than”, “greater than”, “logical and” or “logical or”. The number of output signals is user-defined.

The comparator function can calculate the out signal from the user provided time function, or from a user provided function of any variable in POLCA-T (i.e., the pressure in some volume cell in the model).

NRC RAI 10-2

For most BWR designs, the feed water control system has an option for three element control, how are similar control systems (with more than one input signal) modeled in POLCA-T?

Westinghouse Response to NRC RAI 10-2

A three element controller can be simulated in many different ways by POLCA-T, as shown below:

1) As [ ]$^{a,c}$ several input signals according to the Figure below:

a,c

2) As several [ ]$^{a,c}$ with the output from the controller [ ]$^{a,c}$ block according to the Figure below:

a,c

3) Using the [ ]$^{a,c}$ system simulation.

NRC RAI 11-12

Section 14.4:

(1) Again, cladding reaction with coolant is alloy specific. Please appropriately identify any and all alloys.

(2) For each alloy identified in (1) immediately above, please verify the validity of this section's equations versus experimental data.

(3) Please justify why the Baker-Just model is adequate for POLCA-T.

(4) Is the Cathcart-Pawel model not applicable or necessary?

(5) The first sentence of this section refers to Baker-Just; the second to last sentence of this section refers to Cathcart-Pawel for values of constants. Please clarify.

(6) Please explain why cladding thermal properties do not change as oxide layers develop.

Westinghouse Response to NRC RAI 11-2

1) POLCA-T has the flexibility to model any type of cladding material as long as the appropriate material property correlation is available and implemented. Current code applications mainly utilize [ ]$^{a,c}$.

   Using the code's input data the user has the freedom to describe the fuel rod design and to separately provide the material type for each fuel assembly loaded in the core. POLCA-T utilizes the same models and data already developed and validated in existing Westinghouse fuel rod design methods and codes. [ ]$^{a,c}$

2) The plot in Figure 1 from Reference 1 shows that the reaction rate for total oxygen uptake is clearly higher for Baker-Just than for other expressions all of which are in good agreement with data.

   It can be noted that the oxidation kinetics at high temperatures (more than 900°C) are barely dependent on the alloy type. For operating temperatures (about 300°C) the oxidation rate varies considerably depending on alloy type and on water-chemical conditions; however, these differences are only important if they alter the initial conditions for transients. For high temperature transients only the high temperature oxidation kinetics are of interest.

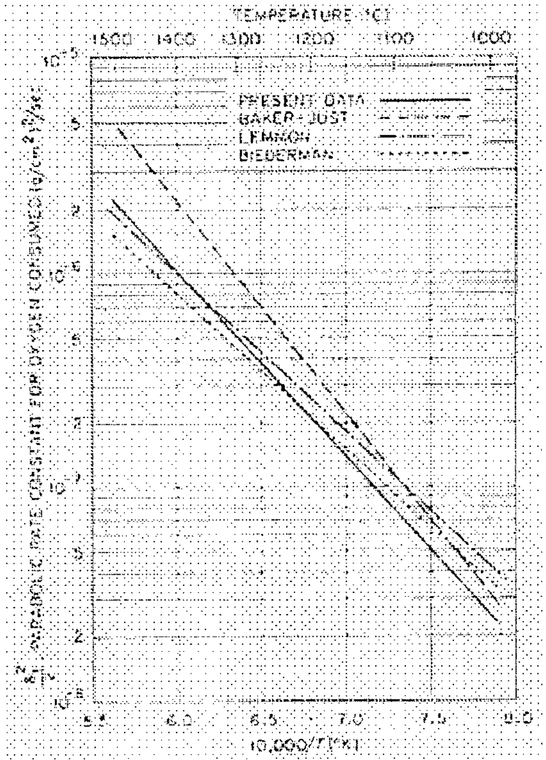


Figure 1, Comparison of Zirconium Metal-Water Oxidation Kinetics

3) In current POLCA-T applications the Baker-Just model [ ]$^{a,c}$ calculation of power generation in the cladding due to metal vapor steam reaction.

4) The Cathcart-Pawel model is available as [ ]$^{a,c}$

5) The Baker-Just correlation is the [ ]$^{a,c}$. For [ ]$^{a,c}$ the Cathcart-Pawel model can be used.

6) The thermal properties change because of the formation of Zirconium oxide and the metal thickness decrease of the cladding.

   This will affect the overall thermal conductance and the heat capacity of the cladding. However [ ]$^{a,c}$ as the cladding oxidation is affected by many factors and most of them are plant specific. As briefly mentioned in 1) the user may supply the code with [ ]$^{a,c}$.

   Thus the use of cladding oxide thickness and oxidation dependent material properties is [ ]$^{a,c}$ of their availability and correctness.

## Reference

[1] ORNL/NUREG-17, "Zirconium Metal-Water Oxidation Kinetics IV. Reaction Rate Studies", J.V. Catcart, R.E. Pawel, et al

NRC RAI 11-13

Section 14.5:

(1) Please explain the values for "a" and "b" in equation (14-84) which are taken from Reference 14.8. Why are they appropriate?

(2) Please explain the data fit in the alpha region.

(3) Justify the linear interpolation in ln(a) and b when used in the lower and upper halves of the $\acute{\alpha}+\beta$ region.

(4) For the sentence, "The burst stress for the double layer has been determined from a data point in Reference 14.4 that implicitly gives the value 113 MPa at 1170°C and assuming the same decay constant (b) as for 3-phase zircaloy."

   (a) Please identify the data point.

   (b) Explain if the data point is justifiably used because it is one point in a well obtained data set.

   (c) Since this is a double layer, why is only the β-phase constant value for "b" assumed?

   (d) Does not the constant "b" vary between alloys?

   (e) To which alloys is this application of the constant "b" being made?

Westinghouse Response to NRC RAI 11-13

1) The reference identification is misprinted. It should be Reference 14.6 (Erbacher et al.). The formulation in Reference 14.6 is based on test data and adopted by [ ]$^{a,c}$. See also the answer to 2) below.

2) The burst stress is compared in Reference 14.6 with data obtained in the REBEKA tests. The Figure below is taken from Reference 14.6 and the [ ]$^{a,c}$.

   The comparison with the REBEKA data is complicated by the influence of oxygen uptake. However in the low temperature area (alpha-region) the amount of oxygen is small and the formulation in Reference 14.6 seems to follow the upper bound of the data points [ ]$^{a,c}$.

   One should bear in mind that rod bursts occur when the true stress equals the burst stress and rod bursts at these temperatures tend to occur as a result of the deformation leading to increased true stress when the wall thickness gets thinner with increasing rod diameter.

a,c

[ ]

3) The interpolation method used is a reasonable way to obtain a smoothly varying burst stress. It is adopted from Reference 14.6.

4) The reference identification is misprinted. It should be Reference 14.8 (Jonsson et al.).

   (a) *Please identify the data point.*

   The data point used is a temperature ramp at [ ][a,c] overpressure. The ramp was designed to reach [ ][a,c] but ruptured at about [ ][a,b,c]. The test is numbered 81-11 in Reference 14-8. A repeat test numbered 81-12 with the same conditions did not rupture. The burst stress of the double layer was determined [ ][a,c] with calculated thickness of the double layer and with calculated burst stress and thickness of the metal.

   (b) *Explain if the data point is justifiably used because it is one point in a well obtained data set.*

   One point was used to set the burst stress of the double layer. All measured points are used to compare with calculations. Only some tests in Reference 14.8 lead to tube rupture but they are all of equal value in the comparison of the model with the data.

   (c) *Since this is a double layer, why is only the beta-phase constant value for "b" assumed?*

   It was postulated that b = 0.003 could be used for the double layer. Then comparison with all the data from Reference 14.8 showed that [ ][a,c]. The rupture temperature in the Reference 14.8 data varies from [ ][a,c] which gives [ ][a,c] value of b for the double layer.

(d) *Does not the constant "b" vary between alloys?*

The data used to compare with the model are obtained mainly with Zircaloy-2 and Zircaloy-4 materials. The data does not indicate any noticeable effect of alloy type. The high temperature tests of Reference 14.8 were made using Zircaloy-2. The oxidation kinetics of various Zirconium alloys at high temperature are believed to be quite uniform and therefore it is reasonable to assume that [ ][a,c].

(e) *To which alloys is this application of the constant "b" being made?*

See answer to (d) above.