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Our ref: LTR-NRC-10-7
February 9, 2010

Subject: Response to the NRC's Request for Additional Information By The Office of Nuclear Reactor Regulation For Westinghouse Electric Company Topical Report WCAP-16498-P "17x17 Next Generation Fuel Reference Core Report" (TAC NO. MD8540) (Proprietary/Non-Proprietary)

Enclosed are copies of the Proprietary and Non-Proprietary versions of the responses to the NRC's Request for Additional Information By The Office of Nuclear Reactor Regulation For Westinghouse Electric Company Topical Report WCAP-16498-P "17x17 Next Generation Fuel Reference Core Report."

Also enclosed is:

1. One (1) copy of the Application for Withholding, AW-10-2747 (Non-proprietary) with Proprietary Information Notice and Copyright 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 Proprietary Information 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-10-2747 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: E. Lenning, NRR
A. Mendiola, NRR
J. Kaizer, NRR
Y. Orechwa, NRR

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Our ref: AW-10-2747
February 9, 2010

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-NRC-10-7 P-Enclosure, "Response to the NRC's Request for Additional Information By The Office of Nuclear Reactor Regulation For Westinghouse Electric Company Topical Report WCAP-16498-P '17x17 Next Generation Fuel Reference Core Report' (TAC NO. MD8540)" (Proprietary)

Reference: Letter from J. A. Gresham to Document Control Desk, LTR-NRC-10-7, dated February 9, 2010

The Application for Withholding Proprietary Information from Public Disclosure 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-10-2747 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-10-2747 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: E. Lenning, NRR
A. Mendiola, NRR
J. Kaizer, NRR
Y. Orechwa, NRR

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

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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 9th day
of February, 2010.

Renee Giampole

Notary Public

COMMONWEALTH OF PENNSYLVANIA
NOTARIAL SEAL
Renee Giampole, Notary Public
Penn Township, Westmoreland County
My Commission Expires September 25, 2013

(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-10-7 P-Enclosure, "Response to NRC's Request for Additional Information By The Office of Nuclear Reactor Regulation For Westinghouse Electric Company Topical Report WCAP-16498-P '17x17 Next Generation Fuel Reference Core Report' (TAC NO. MD8540)" (Proprietary), for submittal to the Commission, being transmitted by Westinghouse letter (LTR-NRC-10-7) and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse Electric Company includes responses to NRC request for additional information.

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

(a) Obtain generic NRC licensed approval for the Westinghouse 17x17 Next Generation Fuel (17x17 NGF) design.

(b) Assist customers in improving their fuel (zero defects).

Further this information has substantial commercial value as follows:

(a) Westinghouse can use this Reference Core Report to further enhance their licensing position over their competitors.

(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.

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).

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Response to NRC's Request for Additional Information By The Office of Nuclear Reactor Regulation For Westinghouse Electric Company Topical Report WCAP-16498-P, "17x17 Next Generation Fuel Reference Core Report" (TAC No. MD8540) (Non-Proprietary)

Westinghouse Electric Company
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355

Request for Additional Information
By The Office of Nuclear Reactor Regulation
For Westinghouse Electric Company Topical Report
WCAP-16498-P "17x17 Next Generation Fuel Reference Core Report"
TAC NO. MD8540
Project NO. 700

PART 1

Westinghouse states that the 17x17 Next Generation Fuel (NGF) design results in very small differences in nuclear design characteristics compared to those of the 17x17 Robust Fuel Assembly (RFA) fuel design. This statement only makes sense in the context of two reference cores, wherein only the fuel assembly designs differ. The following Request for Additional Information (RAI) request a one-to-one comparison between methodologies, computer codes and specific performance characteristics as they relate to the fuel performance analysis of 17x17 RFA and 17x17 NGF fuel designs described in Table 2-1 in WCAP–16498–P *in the context of the same reference core.*

RAI-1: Core Nuclear Design

In the analyses of the neutronic performance of the two designs (i.e. 17x17 RFA vs. 17x17 NGF), what is used for the following analyses. If they differ between the two designs, describe the differences and their consequences.

Cross section generation:

1. What were the basic cross section data? (For example, ENDF/B – Version?) If they differ what are the key isotopes that introduce differences in the results?

Response: The basic cross section data Westinghouse uses in its core design computer codes does not depend on the type of fuel modeled. The current Westinghouse codes use basic cross section data based by ENDF/B-VI.

2. Were the same cell codes and homogenization techniques used for the two designs?

Response: The cell and homogenization techniques Westinghouse uses does not typically depend on the type of fuel modeled. The same codes and techniques will be used for the 17x17 NGF fuel modeling as are currently used for our other fuel types.

3. What and how were the assembly design differences taken in account in the generation of the cross sections?

Response: The only difference that affects neutronics between 17x17 NGF and 17x17 RFA fuel are very slight differences in the volumes of the grid material used. The pellet material, design, and geometry are unchanged. The clad material, design, and geometry are unchanged. The fuel rod and assembly pitch are unchanged.

The differences in material volumes associated with the different grid design for 17x17 NGF fuel are explicitly modeled in the Westinghouse core design codes. Because zircaloy materials are weak neutron absorbers, and because the differences in zircaloy volume are small between the designs, the resulting impact on k-infinity and fuel cross sections is very small.

Neutronic analysis:

1. Westinghouse states the differences in the nuclear design characteristics between the two assembly designs are very small. What is the reference core which allows you to come to that conclusion? How is small defined?

Response: The conclusion that the differences in nuclear design characteristics are very small results from the fact that the design variables that control neutronic performance did not change with the 17x17 NGF fuel design. This is shown in Table 2-1 of WCAP-16498-P. The pellet and clad geometry are identical. The fuel rod and fuel assembly pitch have not changed. ZIRLO™ and Optimized ZIRLO™ are neutronically equivalent. The only difference that affects neutronics between 17x17 NGF and 17x17 RFA fuel are very slight differences in the volumes of the grid material used. [

]a, c.

A comparison of k-infinity as a function of burnup between 17x17 RFA fuel and 17x17 NGF fuel is shown in Figure 1. The assembly modeled was enriched to 4.95 w/o and contained no burnable absorbers in this example. The resulting k-infinity values are so similar that it is not possible to detect any difference from the plot – all three curves fall on top of each other.

Figure 1
K-Infinity versus Burnup Comparison

a, b, c

2. What neutronic codes were used for steady-state, transient and burnup neutronic analyses to generate the required input for fuel performance and system accident analyses?

Response: The standard Westinghouse licensed core design codes were used for both the 17x17 RFA and 17x17 NGF core design analysis. The following references apply:

Liu, Y. S., et al., "ANC - A Westinghouse Advanced Nodal Computer Code," WCAP-10965-P-A (Proprietary), December, 1985.

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

3. Compare the axial nodal mesh in the neutronic analysis of the two assemblies? How does it relate to Span # in Fig. 2-1 of the Topical Report (TR)?

Response: The same number of axial mesh is used. 24 uniform axial meshes are typically modeled in the ANC code. The grids are typically homogenized into the moderator region in the lattice code. However, if grids are modeled explicitly in the 3D neutronic code, then the axial meshes are adjusted to accommodate the actual grid locations and non-uniform axial meshes would be used.

4. What are the material volume fractions within each axial node? How is the difference in the structural material between the two designs taken into account?

Response: The grid materials and volumes associated with a specific design are modeled in the lattice codes by adding the appropriate number density of grid materials to the moderator region of the models. The grids are not modeled explicitly but are smeared in the moderator. This does not change with 17x17 NGF fuel, but the grid volumes for 17x17 NGF are explicitly modeled in the lattice code.

5. What are the limiting power peaking factors: Core - axial and total, Peak assembly - axial and total? Do these peaking factors occur in the same assembly at the same fuel pin in the reference core for the two designs? When in the residence time of the assembly do they occur?

Response: The value and time in life for the core peaking factors depends on the specific core design and fuel loading pattern where the fuel assembly is used. Because of the neutronic similarity between 17x17 NGF and 17x17 RFA, Westinghouse would expect the same core power distribution for either fuel assembly if the core loading pattern was the same. The use of 17x17 NGF fuel does not require changes to core loading patterns or fuel management strategies compared to 17x17 RFA2 fuel.

6. What are the peak and average discharge burnups? Do they occur in the same assembly and fuel pin?

Response: 17x17 NGF fuel will be constrained to the licensed lead rod burnup limit, until a higher value of lead rod burnup is licensed. That value is currently 62,000 MWD/MTU. There is no corresponding licensing constraint on average discharge burnup, but the lead rod burnup constraint generally results in region average discharge burnups not exceeding 50,000 to 55,000 MWD/MTU, depending on the details of the fuel management for an individual plant.

7. What is the beginning of life (BOL), end of life (EOL) and limiting moderator coefficient for each design?

Response: Because the pellet, clad, and fuel assembly geometry is unchanged, and the resulting assembly reactivity is nearly identical between 17x17 NGF and 17X17 RFA fuel, no change in moderator temperature coefficient will occur between a 17x17 NGF core and a 17x17 RFA2 core using the same loading pattern and fuel management.

8. What is the BOL, EOL and limiting rod worth for N-1 shutdown (i.e. stuck rod) for each design?

Response: Because the pellet, clad, and fuel assembly geometry is unchanged, no change in control rod worth or shutdown margin will occur between a 17x17 NGF core and a 17x17 RFA core using the same loading pattern, fuel management, and RCCA design.

RAI-2: Fuel Assembly Thermal-Hydraulic Design

1. Briefly describe the thermal-hydraulic codes, methods that based on the neutronic analyses outputs are used to compute the necessary inputs for the fuel assembly and fuel pin mechanical performance analyses.

Response: The current thermal-hydraulic code and method that interface with Westinghouse neutronic code for the 17x17 NGF are Westinghouse's version of the VIPRE-01 code and the Revised Thermal Design Procedure. They are the same as those used for other 17x17 fuel designs.

WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989.

2. Compare the axial ΔP distribution in the peak channel for the two designs in steady-state operation.

Response: A comparison of the axial ΔP distribution from the hot channels of the 17x17 RFA and the 17x17 NGF fuel designs, as calculated using Westinghouse's version of the subchannel code VIPRE-01, is shown in Figure 2. The comparison was made at the following design condition:

Core Pressure	=	2400 psia,	Core Inlet Flow	=	15.4 ft/s
Core Inlet Temperature	=	546.4 °F	Core Average Power	=	71.0 kW/rod

The axial ΔP distributions at steady state operations would be similar to those in Figure 2.

Figure 2

Comparison of Axial Δ-P Distribution in Hot Channel (17x17 RFA and 17x17 NGF)

a, b, c

3. Compare the axial peak cladding temperature distribution of the peak pins in the cores in steady state operation. Are they the same pin in the two cores?

Response: The axial peaking cladding temperature distribution is dependent mainly on the pin power, the fuel rod heat transfer model, and the cladding material. [ ]$^{a, c}$, there is no difference in the cladding temperature distribution as predicted by the fuel performance code for input to the safety analysis.

4. Compare the peak sub-channel axial coolant temperature distributions in steady state operation.

Response: A comparison of the peak sub-channel axial coolant temperature distributions under the design core condition, as predicted by the sub-channel code VIPRE-01, shows no significant difference between the 17x17 RFA and the 17x17 NGF.

5. What are the maximum net upward forces on the fuel assemblies due to axial flow at steady state operation?

Response: The net upward forces on the fuel assembly due to the axial flow at steady-state operations is a function of the assumed RCS flow rate and to a lesser degree the temperature conditions, which are plant specific. These plant parameters are known for each of the Westinghouse plants where it would be anticipated where the NGF design would be used. For the purposes of evaluating the NGF product design, the lift forces were calculated at anticipated plant design conditions. For the cold last pump start conditions, the maximum lift force is estimated to be less than [ ]$^{a, c}$ and for hot full power conditions, the maximum lift force is less than [ ]$^{a, c}$.

6. What is the figure of merit for assessing the margin to the density wave instability? What are the thermal-hydraulic variables that are used to compute this figure of merit? Which one is the dominant contributor to this particular analysis? What are the values of this figure of merit in the case of the two cores under consideration?

Response: Design basis, acceptance limit and evaluation method for density wave instability are described in following references. Margin to the instability could be affected by a change in fuel pressure drop, and therefore is evaluated for the new fuel design. As noted in WCAP-16498-P, Section 4.1.3, "for Westinghouse reactor designs, the Ledinegg instability is prevented because the slope of the reactor coolant system pressure drop versus flow rate curve is larger than the loop supply (pump head) pressure drop versus flow rate curve. The 17x17 NGF design will maintain the same characteristic of the reactor design" (i.e., the fuel pressure drop difference between RFA and NGF is far less than what is need to cause a density wave instability).

"The margin to the density wave instability will be evaluated using the method of Ishii for the 17x17 NGF design, which is the same as for other Westinghouse fuel designs. An inception of this type of instability would typically require increases on the order of 100% or greater of rated reactor power. Margin to the instability will be verified for each plant application with plant specific core operating conditions."

WCAP-12488-A, "Westinghouse Fuel Criteria Evaluation Process," October 1994.

Saha, P., et al., "An Experimental Investigation of the Thermally Induced Flow Oscillations in Two-Phase Systems," Volume 1, Heat Transfer, November 1976, pp. 616-622.

7. What is the limiting predicted channel closure due to rod bowing for the two cores in steady state? What is the estimate for the limiting transient?

Response: Channel closure due to rod bowing is a function of fuel burnup in the core, as described in the following references. In the fuel region containing an IFM grid, there is no rod bow DNBR penalty because the predicted channel closure is less than 50%. Since the NGF design has two IFM grids in a span between two mixing vane grids, the maximum channel closure is further reduced and the rod bow DNBR penalty is estimated to be zero for DNB limiting transients.

WCAP-8691 Revision 1, "Fuel Rod Bow Evaluation," July 1979.

WCAP-10444-P-A, "VANTAGE 5 Fuel Assembly Reference Core Report," September 1985 (IFM Rod Bow).

RAI-3: Fuel Assembly Mechanical Design

Fuel assembly growth:

1. Describe the codes that compute the required axial clearance between the core plates and nozzle end plates that are used to demonstrate sufficient margin for fuel assembly and fuel pin growth to design burnup in steady state operation. What physical phenomena are taken into account in computing growth? What is the dominant phenomenon in steady state operation?

Response: There are no codes that are specifically used in determining the fuel assembly growth and the required axial clearances as the method used is a straight forward approach which combines different known effects, such as the guide thimble tube growth, the fuel rod growth, and fuel assembly growth data gathered during Post Irradiation Examination (PIE) inspections. The method of how the clearances are determined was first describe in WCAP-9500-A and is re-iterated below for convenience.

Design Basis:

The length of the fuel assembly is established so that it is consistent with the space available between the upper and lower core plates (accounting for thermal expansions). The length is established to prevent axial interference (hard contact between the top nozzle and the upper core plate) from developing due to thermal expansion and irradiation-induced growth of the fuel assembly. If interference were to develop, it could lead to bowed fuel assemblies, guide thimble distortion which could cause possible control rod insertion issues, increased axial loads through the fuel assembly and/or on the core internals, and possible fuel assembly structural failures. Thus, the design requirements for the fuel assembly are that there must be sufficient space for growth due to thermal expansion and irradiation effects throughout the fuel assembly life.

Method:

The method used to determine the required clearances between the fuel assembly and the upper core plate is performed assuming cold/room temperature dimensions. This results in the most conservative clearances since the thermal expansion for the stainless steel associated with the reactor vessel and the internals is greater than the thermal expansion for the fuel assembly, Zircaloy-4/ZIRLO™ based materials. The specific assumptions that are used in this determination are listed below.

- The distance between the lower and upper core plates at cold conditions, including the most conservative direction for the tolerance, is assumed. It should be noted that plant specific measurements are performed to confirm that the actual distances at various locations about the circumference of the upper and lower core plates are well within the design tolerances. The design distance and tolerances are assumed in determining the required clearances. These measurements are known for all Westinghouse NSSS plants but are confirmed by the utility when new fuel designs are first implemented, thus ensuring the documented values are accurate (i.e., any changes that the plant may have implemented are accounted for).

- The fuel assembly growth as a function of irradiation/burnup is based on extensive fuel assembly irradiated growth experience for assemblies with Zircaloy-4 grids and guide thimbles and data from assemblies with ZIRLO™ grids and ZIRLO™ guide thimbles. The fuel assembly growth is determined based on Post Irradiation Examination (PIE) inspections in which measurements are made at various locations about each fuel assembly to determine a bounding growth value. Using this plant data, a conservative upper bound fuel assembly growth curve was established and licensed with the NRC as part of the ZIRLO™/Opt ZIRLO™ topical reports. It should be noted that because of the force applied by the top nozzle springs, the irradiation-induced growth is minimized when compared to the irradiation-induced growth of the fuel rods.

- In defining the fuel assembly distances, tolerances are taken in the most conservative direction in order to determine the minimum required clearances.

With respect to the fuel rod growth method to determine the minimum clearances between the fuel rod and the bottom of the top nozzle, a conservative fuel rod growth model is used based on measured data from actual in reactor operational experience. Again, this is discussed and licensed with the NRC as part of the ZIRLO™/Opt ZIRLO™ topical reports. This conservative fuel rod growth model is used along with the lower bound fuel assembly growth curve to determine if there is sufficient clearance between the fuel rods and the top and bottom nozzles. A nominal differential thermal expansion between the fuel rod cladding and the fuel assembly structure is accounted for as the average temperature of the fuel cladding is greater than that for the guide thimble tubes (fuel assembly structure)..

In steady-state operation, with no changes to the operating temperatures, the most dominant phenomenon with respect to fuel assembly and fuel rod growth would be due to irradiation-induced growth.

2. At what point in the residence time of the fuel are the transient contributions computed? Why are they limiting at that point? What is the dominant phenomenon that contributes to axial growth during the transient at this point in the residence time?

   Transient contributions are generally limited to only the thermal expansion of the fuel in the Westinghouse methodology, which is independent of time of life in the core. When computing transient concerns, such as the peak rod internal pressures during a Loss of Coolant Event, the steady-state fuel cladding axial growth is considered as well as the transient thermal expansion of the core, which is independent of the residence time in the core.

Fuel assembly hydraulic stability:

1. The basis for demonstrating that fuel rod wear due to contact with mid-grids and IFM-grids is well within the Westinghouse guideline for limiting wear is demonstrated by flow tests at the VIPER Loop and at the FACTS Loop. Are the test assemblies at each loop BOL assemblies. If so, how are the radiation and the variation in the thermal-hydraulic environment taken into account. If some or all are neglected, what is the rational and how does this still justify the claim that the test represents the limiting conditions?

Response: With respect to fuel rod wear, Westinghouse has used the Vibration Investigation and Pressure-drop Experimental Research (VIPER) test since approximately 2000, as a means of performing comparative testing of different fuel designs. Tests of reference designs with known in-reactor grid-to-rod fretting (GTRF) performance provide benchmarks for comparison of new grid/fuel assembly designs, such as the 17x17 Next Generation Fuel (NGF) assembly design. [

] [a, c]. These non-oxidized rod wear scar distributions allow for the comparison of different fuel assembly designs. Over the years, the VIPER testing has repeatedly ranked the relative fretting performance of fuel assembly designs with that observed in the field. The intent with any new design, such as the 17x17 NGF assembly design, is to ensure that new designs have better GTRF performance than the reference designs at the same test conditions.

As mentioned above, the VIPER testing does consider irradiation effects such as grid spring relaxation, clad creep-down, grid growth, as well as grid oxidation, etc. in order to simulate an EOL condition, which is limiting with respect to fuel rod wear, although un-irradiated fuel assemblies are used in the testing.

The testing incorporates numerous assumptions and conservatisms, as explained above, to simulate an EOL condition, which is limiting with respect to fuel rod wear. For instance, [

] [a, c]. Post Irradiation Examinations (PIEs) exams have been performed over the years and most notably for the Robust Fuel Assembly (RFA and RFA-2) and NGF assembly designs and have shown that the amount of fuel wear on a 95/95 basis is less than 10% and more importantly there have been no leakers to date, due to GTRF. The RFA PIE exams have included a range of different plant types, fuel configurations (RFA and RFA-2, 12 ft and 14 ft, with and without IFMs), cycle lengths, and operating histories. The RFA-type fuel has operated in 38 plants worldwide with more than 11,000 assemblies and approximately 3 million fuel rods with no leakers. Recently performed PIE exams for the Lead Test Assemblies of the NGF assembly design have also shown that there have been no leakers. Given that the VIPER testing is a comparative test and has shown that the NGF assembly design performs better than the RFA-type design, it is expected that the NGF design will out-perform the RFA design with respect to GTRF. This is shown by Figure 3 which shows the scar extrapolation for the different fuel types to the critical volume [ ]a, c from the VIPER tests. Thus, while not every plant operating condition is specifically captured, the overall VIPER comparative approach, along with PIE exams, is a conservative method of ensuring that the fuel rod wear is minimized thereby reducing the potential for fuel failures due to GTRF.

Figure 3
Comparison of Wear Scar Extrapolation to the Critical Volume

[ ]a, b, c

2. What are the margins to the Westinghouse guideline value for the two cores?

Response: VIPER tests were performed with two test assemblies; a 17x17 NGF assembly with five Intermediate Flow Mixing (IFMs) grids and a 17x17 Robust Fuel Assembly (RFA) with three IFMs. This test simulates what is expected to be a typical NGF transition core environment of the RFA or Vantage 5H fuel product to the 17x17 NGF product. Because the NGF grids have a higher pressure loss than the RFA grids and the NGF assembly contains two additional IFM grids, there is local crossflow that is generated at the NGF grid locations. [

]a, c. Thus, the long-term wear test provided conservative axial flow for both assemblies in either a transition or non-transition NGF core. The results of the VIPER tests indicated that the wear performance for both assembly types met the criterion of less than 10%, with the NGF product demonstrating better wear performance than the RFA fuel product. Again, as a relative comparison, Figure 3 presents the wear scar extrapolation for the different fuel types to the critical volume [ ]a, c from the VIPER tests. This plot shows the improved performance of the NGF with IFMs product in relation to other fuel products.

Based on the above, the NGF design significantly increases the fuel assembly fretting margin while not adversely impacting the fretting behavior of the co-resident RFA fuel in a transition core.

Fuel assembly structural integrity, and shipping and handling loads:

1. What are the Westinghouse figures of merit with regard to seismic and loss-of-coolant accident (LOCA) loads in the case of fuel assembly structural integrity loads, and to what fuel assembly components are they applied?

Response: The key figures of merit with regard to the seismic and LOCA loads include the lateral grid crush strength and the grid stiffness as well as the lateral and axial fuel assembly stiffness. These figures of merit are used in the calculation of the maximum grid impact forces for the fuel assembly grid components, based upon plant specific core plate motions resulting from the Seismic and LOCA events and based upon an analytical model of the fuel assembly rows in the core.

2. Are the margins determined purely by testing; or are some based on computation?

Response: The lateral grid crush strengths are experimentally determined by finding the collapse load for the grid at elevated temperature conditions which are intended to simulate reactor operating conditions. The grid impact forces are calculated using models which simulate the core plate motions resulting from Seismic and LOCA events and based on an analytical model of the fuel assembly rows in the core.

3. What are the margins for each core?

Response: The margins to the lateral grid crush strength are based on plant specific Seismic/LOCA analyses and plant specific models for the reactor vessel and the core.

4. Are the radiation history and the evolution of the thermal-hydraulic environment taken into account in the test and analysis results?

Response: [

]a, c.

Structural components:

For the structural components listed in section 2.4 of the TR:

1. What are the Westinghouse figures of merit with regard to each component?
2. Are the margins determined purely by testing; or are some based on computation?
3. What are the margins for each core?

4. Are the radiation history and the evolution of the thermal-hydraulic environment taken into account in the test and analysis results?

Response: In responding to this question and its associated sub-parts, the approach is to identify the key/important figures of merit for each of the structural components that have changed from prior fuel designs. This approach therefore focuses on specific testing and analyses that were performed to support the new NGF fuel design.

<u>Bottom Nozzle (Section 2.4.1):</u>

1. What are the Westinghouse figures of merit with regard to each component?

The bottom nozzle serves as the bottom structural element of the fuel assembly and distributes the RCS coolant flow to the fuel bundle region of the fuel assembly in addition to helping ensure that debris doesn't reach the active fuel bundle region. The bottom nozzle also prevents the accidental downward movement of the fuel rods from the fuel assembly. The key figure of merit for the bottom nozzle is to ensure that the hydraulic pressure loss is minimized (as noted in Section 2.4.1 of WCAP-16498-P) as this can affect the overall RCS loop pressure drop which affects the RCS flow. Because of the additional IFMs, the NGF bottom nozzle has been optimized to minimize the pressure drop. An increase in the fuel assembly pressure drop can reduce the RCS flow which is non-conservative for safety analyses, although there would have to be a significant increase in the bottom nozzle pressure drop to affect the RCS flow since the bottom nozzle makes up a small part of the overall pressure drop.

Additional figures of merit, as noted in Section 2.4.1, include the load-deflection characteristics and flatness requirements of the nozzle, as well as confirming that the applicable stress limits are met for Condition I and II events, Condition III and IV events and for shipping and handling load conditions.

2. Are the margins determined purely by testing; or are some based on computation?

For the key figure of merit of the hydraulic loss coefficient, margin is not specifically calculated as the intent is to minimize the hydraulic loss coefficient. While computational modeling of the bottom nozzle is sometimes performed, the hydraulic loss coefficient for use in licensing basis analyses is primarily based on testing performed in the Fuel Assembly Compatibility Test System (FACTs) test loop.

With respect to the load-deflection characteristics and flatness requirements, specific tests are run which demonstrated that the applicable requirements were satisfied. The plate flatness requirements are to ensure that the deflection is less than [ ]$^{a, c}$. The Condition I and II event design loads are bounded by the Shipping and Handling Loads. Mechanical testing was performed for the Condition III and IV structural loads. These tests demonstrated that there was no loss of structural integrity, that is, no unrestrained or excessive deformation, cracking, etc. The Shipping and Handling loads were examined assuming a 4g acceleration acting in the axial direction and a 6g acceleration acting in the lateral direction. The testing demonstrated that the applicable criteria, including the top plate deflection and associated stresses in the bottom nozzle were met.

3. What are the margins for each core?

As noted above, for the figure of merit of the hydraulic loss coefficient, there is no margin per se'. However, there is margin in the RCS flow which is directly dependent on the RCS loop resistance, although the bottom nozzle is a very small portion of the RCS loop resistance, a change in the bottom nozzle loss coefficient can affect RCS flow margins. Since the RCS flow varies from plant to plant, the RCS flow margins also are plant specific. However, the flow margins are know for each of the Westinghouse plants where it would be anticipated where the NGF design would be used and based on the known increase this design would have on the RCS loop resistance, it is known that this change is acceptable for all of these plants.

With respect to shipping and handling load requirements, margins are not specifically defined per se' but rather it is confirmed that the applicable acceptance criteria are satisfied. For the top plate deflection, the largest residual deflection after the test loading was [

]$^{a, c}$.

4. Are the radiation history and the evolution of the thermal-hydraulic environment taken into account in the test and analysis results?

The bottom nozzle hydraulic loss coefficient is unaffected by radiation effects. The shipping and handling load related criteria are not adversely affected by radiation and thermal hydraulic effects.

Top Nozzle (Section 2.4.2):

1. What are the Westinghouse figures of merit with regard to each component?

The top nozzle functions as the upper structural element of the fuel assembly and provides a partial protective housing for the rod cluster control assembly and other core components, such as guide thimble plugs. The flow slots in the adapter plate permit flow to pass upward through the nozzle and the associated ligaments of the flow slots limit the upward movement of the fuel rods. As noted in WCAP-16498-P, the Westinghouse Integral Nozzle (WIN) top nozzle is functionally the same the current 17x17 designs, with the primary difference being the WIN design eliminates the separate clamp pad and spring screw (nickel alloy threaded fastener) which can be susceptible to primary water stress corrosion cracking (PWSCC). The WIN design, instead, pins the leaf springs into the nozzle corner pad that is machined to capture the spring ends (tails). Since the NGF assembly has additional IFMs, its overall pressure drop is increased when compared to the RFA assembly. Therefore, a 4-leaf spring pack is generally required instead of the more standard 3-leaf spring pack. Because of this specific change, the shipping and handling load stresses for the machined recess in the top nozzle top plate for the spring location were specifically analyzed. Thus, the figure of merit of concern for the WIN top nozzle for the NGF fuel design is stresses at this top nozzle top machined recess location. The remainder of the top nozzle figures of merit remain unchanged, such as the load-deflection characteristics and flatness requirements which must be satisfied, as well as confirming that the applicable stress limits are met, under 4g loads acting axially and 6g loads acting laterally for the shipping and handling load conditions.

2. Are the margins determined purely by testing; or are some based on computation?

To demonstrate that the stresses are acceptable both testing and detailed FEA methods are employed to ensure that the applicable requirements are met.

3. What are the margins for each core?

With respect to stresses at the top nozzle top plate spring location, the safety factor was calculated to be [ ]$^{a, c}$ for loading conditions.

4. Are the radiation history and the evolution of the thermal-hydraulic environment taken into account in the test and analysis results?

Irradiation tends to increase the top nozzle yield strength thereby improving margins and the thermal-hydraulic environment does not adversely affect the top nozzle performance.

<u>Fuel Assembly Hold-down Springs (Section 2.4.3):</u>

1. What are the Westinghouse figures of merit with regard to each component?

The key figure of merit for the fuel assembly hold-down springs is the hold-down force. Detailed calculations are performed which take into consideration the fuel assembly lift forces due to the RCS flow, fuel assembly buoyancy effects, fuel assembly weight, fuel assembly growth, irradiation degradation effects on the springs as well as relative thermal expansion effects of the fuel and core barrel. The criterion for this key figure of merit is to ensure that the hold-down springs provide sufficient hold-down force to ensure that the functional requirements for fuel assembly hold-down is met (fuel does not lift-off during any Condition I or II event other than a hot pump over-speed (turbine over-speed) transient associated with a loss of external load event).

2. Are the margins determined purely by testing; or are some based on computation?

For the key figure of merit for the fuel assembly hold-down springs, the spring hold-down force is based on both detailed testing and analyses. The testing includes FACTs pressure drop testing, which was performed to determine the NGF fuel assembly loss coefficients and for the purposes of determining the fuel assembly lift forces. Extensive plant measurements are collected to determine bounding fuel assembly growth effects, as well as for the purposes of determining the spring stress relaxation due to neutron fluence effects. The analyses include plant specific assumptions for the RCS flow rate, temperatures, and plant specific spring pack designs. Assuming conservative assumptions, analyses are performed to demonstrate that fuel assembly lift off during any Condition I or II event, other than a hot pump over-speed (turbine over-speed) transient, is precluded.

3. What are the margins for each core?

As noted above, the margins for the fuel assembly hold-down spring force are plant specific. Generally, a desired margin of at least 100 lbs is applied in analysis. However, in some instances the margin may be less than 100 lbs. This is deemed to be acceptable based on the many conservatisms and in the fuel assembly top nozzle hold-down spring calculations.

4. Are the radiation history and the evolution of the thermal-hydraulic environment taken into account in the test and analysis results?

As noted above, irradiation and thermal effects are accounted for in the determination of the spring force as well as in the determination of the fuel assembly growth. Generally EOC-2/BOC-3 cold conditions are the most limiting as beyond this point, the fuel assembly growth tends to overpower the effect of irradiation relaxation to the hold-down springs, and, cold flow condition provides the greater lift force against the assembly.

Guide Thimbles and Instrument Tube (Section 2.4.4):

1. What are the Westinghouse figures of merit with regard to each component?

The guide thimbles are the structural members of the fuel assembly which also provide channels for the insertion of neutron absorber rods, burnable poison rods, neutron sources, or guide thimble plug assemblies. The key figure of merit for the guide thimbles is to ensure that applicable stress limits are met for the Shipping and Handling loads and for the Condition I and II events. Meeting the stress limits help ensure that the rod cluster control assemblies travel to the fully inserted position thereby ensuring that the reactor is shut down and that the safety analyses remain valid. As noted in WCAP-16498-P, the newer tube-in-tube design is being incorporated into all of the new Westinghouse designs as a means to add significant margin to help preclude incomplete rod insertion (IRI).

2. Are the margins determined purely by testing; or are some based on computation?

The figure of merit of the stresses for the guide thimbles is primarily determined by calculation and does not involve any computer codes.

3. What are the margins for each core?

The guide thimble stress margin is based on the location and the case in question. For the Shipping and Handling loads, the margin is approximately [ ] a, c to the stress limits. For the Condition I and II events, the margin [

] a, c. For the Condition III and IV events, it is required that deflections or deformations of components must not interfere with safe reactor shutdown. This is ensured by demonstrating that the primary and combined stresses from the Condition IV LOCA and Seismic events are less than the applicable limits. The primary stress is [ ] a, c and the combined stress is [ ] a, c. In addition, for Condition III and IV events, it is also demonstrated that the critical buckling load as defined by the ASME Code is not exceeded. A specific calculation was not performed as the 17x17 NGF fuel assembly has the thimble tube-in-tube dashpot design which is improved relative to the 17x17 RFA fuel assembly.

4. Are the radiation history and the evolution of the thermal-hydraulic environment taken into account in the test and analysis results?

No. Irradiation tends to increase the yield strength of the guide thimble thereby improving margins and is thus not credited. In addition, the thermal-hydraulic environment does not adversely affect the guide thimble stresses.

Joints and Connections (Section 2.4.6):

1. What are the Westinghouse figures of merit with regard to each component?

The main function of the joints and connections is to attach the major components of the fuel assembly into a single structure that will maintain a proper geometry for all design conditions. In addition to withstanding the loads involved between the components, these attachments have to be compatible with the thermal hydraulic design of the fuel and core components. Therefore, the key figure of merit for the joints and connections is to ensure that the applicable joint/connection strength is met.

2. Are the margins determined purely by testing; or are some based on computation?

The joint and connection strengths are typically determined via testing.

3. What are the margins for each core?

Generally, the margins are significant. As an example, the top grid sleeve joint strength is greater than [ ]$^{a, c}$. For the mid grid sleeve joint strength, the margin is greater than [ ]$^{a, c}$.

4. Are the radiation history and the evolution of the thermal-hydraulic environment taken into account in the test and analysis results?

No. Irradiation tends to increase the yield strength of the joint and connections thereby improving margins and is thus not credited. In addition, the thermal-hydraulic environment does not adversely affect the guide thimble stresses.

Grid Assemblies (Section 2.4.6):

1. What are the Westinghouse figures of merit with regard to each component?

The grid assemblies perform a number of functions including providing fuel rod support, maintaining fuel rod spacing, forming part of the fuel assembly structure, promoting mixing of the reactor coolant, providing lateral support and positioning the RCCA guide thimble and instrument tubes, helping to prevent damage during handling operations as well as protecting the fuel rods from foreign material in the flow stream.

The key figures of merit for the grids include ensuring that the grid-to-rod fretting is minimized (primarily for the mid-grids) and ensuring that sufficient mixing of the reactor coolant occurs to help ensure that DNB is precluded. The grid-to-rod fretting is discussed in the previous response under Fuel Assembly Growth.

2. Are the margins determined purely by testing; or are some based on computation?

The grid-to-rod fretting margins are discussed in the Fuel Assembly Growth section of the RAI responses. For the DNB performance, testing was performed to generate data to be used in defining a DNB correlation specifically for the NGF fuel product. The DNB correlation was reviewed and approved by the NRC in WCAP-16766-P-A.

3. What are the margins for each core?

With respect to the DNB performance, the data from the testing of the NGF fuel is used to define a DNB correlation specifically for the NGF fuel product and there is no margin created. Usually, a 95/95 probability/confidence level is used in defining the correlation. Again, this was reviewed and approved by the NRC in WCAP-16766-P-A.

4. Are the radiation history and the evolution of the thermal-hydraulic environment taken into account in the test and analysis results?

The DNB performance of fuel assembly/grids is independent of the irradiation history. Thermal-hydraulic environment is specifically taken into account when analyzing for the DNB criterion, as evidence by the analysis of the Condition I and II events as presented in Chapter 15 of the Final Safety Analysis Report for current operating plants.

RAI-4: Lead Test Assembly Program

1. What is measured and what are Westinghouse's limits on these measured values? How do you arrive at the limits for EOL conditions and those for inspections before EOL?

Response: For each of the lead test fuel assemblies, the following Post-Irradiation Examinations (PIEs)/measurements are typically performed.

[ ] a, b, c

In addition to the above, single rod examinations are performed for a subset of fuel assembly rods.

[ ] a, b, c

The limits for EOL conditions are primarily based on historically data collected over the years from numerous plants and for numerous fuel types. Generally the same limits are applied independent of time in life.

2. What is the physical basis for the limiting values, in particular how do you account for the time-dependent irradiation and temperature environment of the fuel assembly in computing the margin?

Response: The limiting values are primarily based on years of data collection which include irradiation and temperature effects as the data is based on actual in-reactor operation. As far as margins, generally new fuel designs are implemented which improve margins to the applicable limits based on a detailed understanding from years of operational experience and PIE exams, as well as testing and analyses.

3. How do you assess the uncertainty in the figures of merit that are compared to the limits?

Response: Generally, the data from the PIEs are used directly. If, for example, it is desired to have a relationship for fuel assembly growth as a function of burnup, the data will be used to define either a best estimate curve, a bounding curve or perhaps a 95/95 basis curve. Additional conservatisms are often applied in other areas when employing actual plant measurement data.

4. Are the lead test fuel assemblies periodically disassembled for inspection and measurement and reconstituted for further irradiation?

Response: Yes, in some cases a once or twice burned lead test assembly is disassembled and examined and then reconstituted for further irradiation.

5. For the test fuel how do you quantify gas pressure and fuel-cladding mechanical interaction? How do you determine the uncertainty in these quantities?

Response: For the test fuel, the rod internal gas pressure (RIP) and the pellet-fuel mechanical interaction (PCMI) were evaluated using the fuel performance code PAD 4.0. The test rods are evaluated using the specific power histories for the fuel cycles in which the test rods operated and accounting for the standard model and manufacturing uncertainties when evaluating RIP and PCMI (clad transient strain and stress).

Selected 17x17 NGF test rods were extracted and measured in the spent fuel pool following the second and third cycles of operation. Among the measurements obtained are diameter and oxide thickness profiles. These measurements characterize the combined interaction of RIP, cladding creep and fuel swelling.

Direct measurement of RIP requires the rods to be taken to a hotcell for measurement. The only poolside measurement of RIP possible is indirectly through the measurement of cladding creep-down. Since after two cycles the burnup of the 17x17 NGF test rods was enough to result in pellet clad contact the only determination that can be made with respect to RIP was that it was bounded by fuel rods with similar duty. Comparison of the Optimized ZIRLO™ clad 17x17 NGF test rods with ZIRLO™ rods from the same reactor at similar burnups indicates the same creep down, but lower corrosion for the 17x17 NGF test rods. Creep data on Optimized ZIRLO™ cladding were obtained from other programs and the results have been sent to the NRC in the second reference.

References

WCAP-15063-P-A, Revision 1-A, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," April 2000.

LTR-NRC-08-60, "SER Compliance of WCAP-12610-P-A & CENPD-404-P-A Addendum 1-A 'Optimized ZIRLO™'," (Proprietary/Non-proprietary).

PART 2

RAI-1: Provide the Fuel Rod Design Calc Note.

Response: All calcnotes will be made available for audit purposes, but will not be submitted.

RAI-2: Provide the Thermal-Hydraulic Design Calc Note.

Response: All calcnotes will be made available for audit purposes, but will not be submitted.

RAI-3: Provide the Core Design Calc Note.

Response: All calcnotes will be made available for audit purposes, but will not be submitted.

RAI-4: Fill in the following table for 17x17 NGF Fuel providing the appropriate units (sample units are given).

[ ] a, b, c

a, b, c

Cladding cold work – Cold-work of the cladding (fractional reduction in cross-section area due to processing).

Cladding texture factor – defined as the fraction of cladding cells with basal poles parallel to the longitudinal axis of the cladding tube.

1. The above table will be used to generate the FRAPCON input deck. Currently, FRAPCON does not have Optimized-ZIRLO as a 'cladding type'. Therefore, discuss any differences between ZIRLO and Optimized-ZIRLO for any relevant input parameters (such as 'Cladding cold work').

Response: The differences between ZIRLO™ and Optimized ZIRLO™ have been outlined in the following approved topical reference. The cladding fabrication inputs for FRAPCON-3 should be those appropriate for pRXA material.

Reference: WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A

2. Provide the Radial Fall Off Curve as well as the associated axial power distributions.

Response: Radial Fall Off Curve for Peak Rod

a, b, c

a, b, c

a, b, c

a, b, c

a, b, c

a, b, c

3. Provide the Radial Fall Off Curve for 5 peak rods as well as the associated axial power distributions.

a, b, c

[

]

4. What is the crud deposition rate?

Response: [ ]a, b, c.

5. What is the initial crud thickness?

Response: [ ]a, b, c.