September 10, 2007

Mr. James A. Gresham, Manager
Regulatory Compliance and Plant Licensing
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: ACCEPTANCE FOR REVIEW AND REQUEST FOR ADDITIONAL INFORMATION RE: WESTINGHOUSE ELECTRIC COMPANY (WESTINGHOUSE) TOPICAL REPORT (TR) WCAP-16747-P, REVISION 0, "POLCA-T: SYSTEM ANALYSIS CODE WITH THREE-DIMENSIONAL CORE MODEL" (TAC NO. MD5258)

Dear Mr. Gresham:

By letter dated April 26, 2007, Westinghouse submitted for U.S. Nuclear Regulatory Commission (NRC) staff review TR WCAP-16747-P, Revision 0, "POLCA-T: System Analysis Code with Three-Dimensional Core Model." The NRC staff has performed an acceptance review of the TR WCAP-16747-P, Revision 0. We have found that the material presented is sufficient to begin our comprehensive review. However, the NRC staff has determined that additional information is needed to facilitate the review and has enclosed a request for additional information (RAI) with this acceptance letter. The NRC staff needs to receive answers to these RAI questions by December 31, 2007, in order to support later schedule milestones for the review of this TR.

Section 170.21 of Title 10 of the *Code of Federal Regulations* requires that TRs are subject to fees based on the full cost of the review. You did not request a fee waiver; therefore, NRC staff hours will be billed accordingly.

The NRC staff expects to issue a second RAI by January 15, 2008, and issue its draft safety evaluation by April 30, 2008. The NRC staff estimates that the review will require approximately 500 staff hours, including project management time. The review schedule milestones and estimated review costs were discussed and agreed upon in a telephone conference between Michael Riggs, Principal Engineer, Fuel Engineering Licensing, and the NRC staff on July 30, 2007.

If you have questions regarding these matters, please contact Jon H. Thompson at (301) 415-1119.

Sincerely,

/RA/

Stacey L. Rosenberg, Chief
Special Projects Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 700

Enclosures: 1. Non-Proprietary RAI Questions
2. Proprietary RAI Questions

cc w/o encl 2:
Mr. Gordon Bischoff, Manager
Owners Group Program Management Office
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355
gordon.c.bischoff@us.westinhouse.com

If you have questions regarding these matters, please contact Jon H. Thompson at (301) 415-1119.

Sincerely,
/RA/
Stacey L. Rosenberg, Chief
Special Projects Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 700

Enclosures: 1. Non-Proprietary RAI Questions
2. Proprietary RAI Questions

cc w/o encl 2:
Mr. Gordon Bischoff, Manager
Owners Group Program Management Office
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355
gordon.c.bischoff@us.westinhouse.com

DISTRIBUTION:
PUBLIC (except for enclosure 2 which is non-public)
RidsNrrLADBaxley
RidsNrrDssSnpb
PSPB Reading File
RidsAcrsAcnwMailCenter
RidsNrrDpr
RidsNrrPMJThompson
RidsNrrDprPspb
RidsOgcMailCenter
SRosenberg (Hardcopy)

Package Accession No. ML072420403
Cover Letter and Enclosure 1 Accession No. ML072420372
Enclosure 2 Accession No. ML072420409 *No major changes from input **NRR-106**

OFFICE	PSPB/PM	PSPB/LA	SNPB/BC*	PSPB/BC
NAME	JThompson	DBaxley	AMendiola	SRosenberg
DATE	9/ 7 /07	9/ 7 /07	6 / 13 /07	9/ 10 /07

OFFICIAL RECORD COPY

REQUEST FOR ADDITIONAL INFORMATION

WESTINGHOUSE ELECTRIC COMPANY (WESTINGHOUSE)

TOPICAL REPORT (TR) WCAP-16747-P, REVISION 0,

"POLCA-T: SYSTEM ANALYSIS CODE WITH THREE-DIMENSIONAL CORE MODEL"

PROJECT NO. 700

Request for Additional Information (RAI) 1: Long Cycle Cores

The U.S. Nuclear Regulatory Commission (NRC) staff has several questions regarding the application of POLCA-T to transients initiated from conditions typical of boiling water reactors (BWRs) operating with long cycle durations (i.e., 24 month cycles). Please address the following items.

(1) For long duration cycle core designs, substantial quantities of burnable poisons tend to be loaded in the fuel. In many cases these loadings may exceed seven weight percent Gd_2O_3 in a large number of pins for modern fuel designs. Provide a quantification of biases and uncertainties in pin power peaking, infinite eigenvalue, and Gadolinia loaded pin power as a function of exposure for typical modern fuel designs using the PHOENIX code.

(2) For long duration cycle core designs, black and white control rod patterns may not be representative of the reactor core as operated for modern designs. Demonstrate that the uncertainties established for nodal and global core nuclear parameters established in the approved PHOENIX/POLCA methodology remain applicable to cores with potentially more limiting control rod patterns in a statistically significant manner. Provide specific comparisons and discussion of control blade history effects. Where applicable, provide comparisons to extended cycle plant data. If possible, quantify the uncertainties associated with the control blade history modeling techniques by comparison to traversing in-core probe (TIP) measurements for bundles where the power is suppressed for significant periods of operation (i.e., for leaking fuel bundles).

(3) Quantify any uncertainties in the exposure-dependent fuel rod models in a statistically significant manner in terms of conductivity or gap conductance that are influenced by burnable poison loading. Quantify this uncertainty based on analyses that include only medium-to-high Gadolinia loadings that are representative of modern fuel designs.

(4) Determine the impact of long cycle exposure (24 months) on the calculational efficacy of the code to predict hot and cold eigenvalues. Perform a statistically significant assessment of any uncertainties or biases and describe any design or administrative margin that assures adequate shutdown margin throughout the cycle.

RAI 2: Mixed Cores

The NRC staff has several questions regarding the application of POLCA-T to transient analysis for mixed fuel vendor cores or cores with modern fuel designs. Please address the following items.

(1) Justify the application of the void-quality correlations to modern fuel designs. Consider the range of void fraction and quality where core designs are operated, as in the case of expanded operating domains. The justification should specifically address the applicability of these correlations at high void fraction and to fuel designs with 10 x 10 arrays that include geometric features such as water crosses, boxes, or rods. It should address applicability to radial power shapes representative of heavily burnable poison-loaded lattices (greater than seven weight percent Gd_2O_3 in several rods) under control states typical of operation for long cycle durations.

(2) As discussed on page five of the NRC staff's safety evaluation (SE) for TR CENPD-390-P-A, "The Advanced PHOENIX and POLCA Codes for Nuclear Design of Boiling Water Reactors" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML003730961): "when applying PHOENIX/POLCA to transition cores, ABB/CE should use fuel specific data to model the thermal and hydraulic behavior of the non-ABB/CE fuel and confirm that the uncertainties derived for ABB fuel are applicable to the non-ABB/CE fuel." Provide this information for current fuel designs and for any ABB/CE fuel as operated in modern core designs.

(3) Provide the critical heat flux or critical power correlations used for each fuel design for which approval is sought. Also provide the basis for these correlations for other vendors' fuel designs.

(4) Does the POLCA7 capability to model SVEA fuel as **[** **]** extend to design features common in other vendors' fuel designs?

(5) Describe the pin power reconstruction model as it is applied to other vendors' fuel designs, addressing lattice features such as water rods or boxes and Gadolinia loading patterns.

(6) What, if any, inconsistencies in the transient response are attributed to core cycle analyses performed on a core using a core monitoring system that is designed by another vendor, for instance: 3DMONICORE?

RAI 3: Expanded Operating Domains

The NRC staff has several questions regarding the application of POLCA-T to transient analysis for expanded operating domain BWRs. Please address the following items.

(1) For modern BWR core designs, the core power and flow maps may be extended to include an expanded operating domain, for example extended power uprates (EPUs). Modern core designs with higher numbers of higher power bundles at potentially higher bundle power to flow ratios warrant investigation of the nuclear methods applicability to these domains. Provide a statistically significant assessment of the nuclear design method uncertainties in regards to pin power, bundle power, hot eigenvalue, and void reactivity feedback for expanded operating domain BWR applications. The assessment should include plants operating at Stretch Power Uprate/Increased Core Flow, EPU, Maximum Extended Load Line Limit (MELLLA), MELLLA Plus (MELLLA+), or very high power density conditions. This assessment should address any heretofore unquantified uncertainties in regards to void fraction, burnable poison depletion, hard spectrum exposure accounting, plutonium buildup, and any exposure biases.

(2) Operation in expanded operating domains may include flatter radial power shapes in conjunction with higher powered bundles and lower bundle flows than those included in the qualification of the nuclear design methods in 1999, as documented in TR CENPD-390-P-A. Quantify any potential for excessive bypass void formation as a result of direct moderator heating, or heating of the bypass due to heat released from structures such as the channels or control blades. In light of the quantification, provide justification of the modeling of the bypass flow paths in the methods described in the Appendices to TR WCAP-16747-P. Justify the applicability of nuclear instrumentation models based on the potential for increased bypass voiding relative to the original qualification under steady state or transient conditions.

(3) Quantify uncertainties in nodal parameters and nodal reactivity feedback coefficients as a result of long exposure under hard spectrum conditions. These hard spectrum conditions may arise due to depletion under controlled conditions, or depletion under high void conditions that may be a result of operation in an extended operating domain, or as a result of depletion with unique control rod patterns or burnable poison loadings to extend cycle length. Compare these uncertainties to the uncertainties established in the original qualification basis.

(4) Provide justification of the continued use of the pin and bundle power uncertainties provided in TR CENPD-390-P-A by providing comparative results of analyses using PHOENIX and/or POLCA with comparison to recent gamma scan data for modern fuel designs operated under expanded operating domain conditions.

(5) Provide the NRC staff with qualification of the extension of the constitutive models (i.e., closure relationships) and heat transfer correlations to bundle power and flow conditions that bound those experienced in expanded operating domains. These bounding values should consider exit quality, mass flux, boiling length, exit and average void fraction, and axial power shapes. Confirm that historically reported uncertainties are valid using a statistically significant sample of the population of data used to generate the constitutive correlations.

(6) Include a demonstrate analysis for EPU applications. Provide the results of the analysis in the form of several figures that plot the following bundle operating conditions as a function of exposure for the EPU maximum bundle operating conditions: maximum bundle power, maximum bundle power/flow ratio, exit void fraction of the maximum power bundle, maximum channel exit void fraction, peak linear heat generation rate (LHGR), and peak end-of-cycle nodal exposure. Provide a quarter-core map (assuming core symmetry) showing the bundle operating linear heat generation rate (i.e., maximum LHGR) and the minimum critical power ratio (MCPR) for beginning-of-cycle, middle-of-cycle, and end-of-cycle. Similarly, show the associated bundle powers. When POLCA-T is applied for plant-specific analyses, include this information as a supplement to the plant-specific application.

(7) Provide a discussion of how the core follow data is used to benchmark the analytical methods. Explain the important plant instrumentation readings that are obtained from the plants to simulate the core response using "offline" calculations. Discuss how the data is compared to the core monitoring system predictions. Provide tabulated data, comparing the calculations and the plant's core monitoring system calculational results (e.g., core thermal power, exposure, core flow, thermal limits calculations) for the given cycle data points. Use core follow data from a high density BWR plant operating with the highest core void conditions. Include core follow data for operation in the high power/low flow off-rated conditions for a high power density plant. This is of interest in order to assess the code system’s accuracy under high void off-rated conditions close to the EPU/MELLLA+ conditions.

(8) The objective of this RAI is to determine the accuracy of PHOENIX/POLCA for the current operating strategies (expanded operating domain applications). Select plants with challenging core designs (e.g., uprated plants and high power density plants with extended cycles) to benchmark the codes. The data from the

plants should be statistically significant to current BWR operating strategies and fuel designs. The core tracking cycle exposure should extend to the number of cycles a fuel bundle may remain loaded in a core. Provide plant-specific information for each set of core follow data (the plant type, whether the power level has been uprated, power density, operating domain, fuel type, cycle length, etc.). For each TIP reading, give the cycle state point, the operating power/flow state point, and the corresponding calculated thermal margin available. Evaluate the plant-specific data, including whether the core follow data indicates that the code is less accurate for higher in-channel void conditions. Explain any trends in the data in terms of operation at higher operating domains, cycle length, uprated, and high power density plants. Demonstrate that the current uncertainties and biases used in the analytical method remain valid and applicable.

(9) Several BWRs currently operate with lower core flow ranges at rated power. However, the general practice is to benchmark the codes for plant operation at rated conditions on the assumption that plants do not routinely operate at the lower flow conditions. The low-flow conditions can be limiting for the thermal-hydraulic conditions (e.g., higher void conditions, axial and radial power peaking, and distribution) that adversely affect the performance of the core and the fuel (critical power ratio response). As far as the available data allows, provide a statistically significant assessment of the POLCA-T code suite to model reactor behavior at low core flow off-rated conditions.

(10) Core follow data is based on statistically-averaged values that may not reflect how well the codes predict the conditions in the high-powered bundles. In addition, the core follow TIP readings average out the four-bundle TIP readings axially within the bundle, along with all the TIP readings for a given cycle state point. In some cases, the TIP readings for different cycle points and different sets of core follow data are statistically averaged to determine the uncertainties of the core simulator codes. This approach tends to mask the accuracy of the codes in predicting hot bundle radial and axial power distribution. Using a limiting loading pattern (two or three hot bundles around an instrument string), benchmark the accuracy of codes in predicting the radial and axial power distribution for these four bundles. Include challenging core designs in the hot channel data. Provide the corresponding calculated void distribution for the hot channels.

(11) The objective of this RAI is to determine whether the statistical combination and normalization of the measured and the calculated TIP data comparisons show the axial and nodal differences between the calculated and the measured data for a radial TIP cell. Using a limiting four-bundle TIP cell (limiting number of hot bundles in a control cell, limiting enrichment, limiting cycle exposure point), tabulate the TIP calculated and measured data. Show how the axial, radial, and overall TIP difference, uncertainty, and/or bias is calculated from the TIP

readings. For the same four-bundle TIP data, compare the absolute calculated and measured values for each TIP element reading and provide a tabulation of the corresponding bundle axial void profiles and the absolute difference in TIP data. Evaluate the absolute difference in TIP readings and determine whether the fidelity of the TIP readings varies axially with void. Compare the TIP data with core follow TIP readings for less challenging core and lattice designs and determine whether the bundle power uncertainties increase. Since the four-bundle instrumented cell can contain bundles at different exposures, explain how the accuracy of the methods can be benchmarked for depletion under high-void conditions by using the core follow data. Use gamma scan data, if available, for bundles and peak pins at different exposures (e.g., fresh, once-burned, twice- burned). As an interim measure, select four-bundle TIP readings and cycle state points to assess the fidelity of PHOENIX/POLCA for depletion at high-void conditions. State whether the accuracy of the code for the hot bundle changes with exposure at core conditions as close to EPU or MELLLA+ conditions as possible.

RAI 4: Code Legacy

The NRC staff has several questions regarding the legacy of the constitutive codes that form the basis for POLCA-T. These questions are in regard to clarification of historical information, code usage, and changes that may have been made since the last NRC staff review. Please address the following items.

(1) Provide a list of all code changes made since PHOENIX/POLCA was submitted to the NRC staff for review and approval (1999) that had the potential to affect the implementation of the methodology as approved. Provide a core follow reanalysis of a case contained in the original submittal to demonstrate that changes made since the original review have not resulted in code drift over time.

(2) TR CENPD-390-P-A describes the models in the PHOENIX and POLCA methods. If these models are implemented in the PHOENIX or POLCA codes with adjustments relative to their description in the TRs, provide the basis for those adjustments. If code-to-code comparisons were used to normalize or adjust models in an "empirical" manner, provide the details. An example of this type of adjustment may be to include a numerical bias when using a particular model to improve accuracy and that bias has been changed or added contrary to the verbatim description in the approved methodology TR.

(3) Describe the base case and branch case analyses that are performed with PHOENIX to develop the cross-section input models to POLCA for standard licensing analyses. In cases where these branch cases do not encompass operational parameters provide justification of the extrapolation of nuclear parameters to these values, for example extrapolation in POLCA to **[**

] not analyzed explicitly by PHOENIX. If this is a standard process, include this in an update to TR WCAP-16747-P. Otherwise provide the guidance as stated in internal procedures or manuals for determining the base and branch cases in the update.

(4) Explain how uncertainties in the POLCA-T methodology are combined to determine safety limits. Include this information in an update to TR WCAP-16747-P.

(5) Explain how a bundle specific R-factor is determined.

(6) How is direct moderator heat assigned to internal liquid flow paths (i.e., water cross), internal two phase flows, or external flows in the bypass?

(7) Provide additional information on the procedures for selecting the pump homologous curve input.

(8) Please provide a greater level of detail in the description of the time step size control algorithm(s).

(9) In reactivity transients such as a rod drop the key parameters that should control time step size are the rate of control rod reactivity insertion and the rate of change in fuel temperatures. What steps does Westinghouse take to insure that they get numerically converged results for rod drop transients in light of the fact that the time step control algorithm does not monitor the rate of change in the key controlling parameters in the event?

(10) The NRC staff wishes to perform confirmatory calculations with both the Westinghouse and independent NRC methods. To facilitate this effort, provide the NRC staff with access to the codes used to perform the analyses in the TR. Also provide access to the input decks used for the analyses and the user's manuals.

RAI 5: Individual Models and Separate Effects Qualification

The NRC staff has questions regarding specific models, sensitivities, and separate effects qualifications. Please address the following items.

(1) Provide additional descriptive details of the database used to develop the void-quality correlations. Specifically provide as a separate table the nature of any transient tests performed, the range of pressures, mass flow rates, and heat fluxes tested.

(2) Verify whether PHOENIX is exercised with a 34 group or 89 group neutron cross section library and the Evaluated Nuclear Data File (ENDF) basis for these collapsed libraries. If the 34 group library is used, justify the application of the **[ ]** for non-ABB/CE fuels and modern fuel designs.

(3) If approval is sought for application of POLCA-T to Mixed Oxide (MOX) fueled cores, provide qualification of plutonium depletion effects against a sophisticated Monte Carlo or advanced deterministic transport approach with independent isotopic tracking capabilities for a range of void fraction, loading, and temperatures consistent with modern core designs.

(4) Please evaluate the sensitivity of burnup predictions on MCPR for the transients of interest.

(5) Please evaluate the effects of power distribution uncertainty on MCPR for the transients of interest.

(6) Please discuss the methods, using equations where applicable, that are used to determine the gamma smeared pin power distribution.

(7) Please describe the benchmarks, biases, and uncertainties in the void reactivity coefficient. How are these determined? Once they are determined, how are they implemented in evaluating safety or operational margins?

(8) Please describe the methods used to calculate the non-condensible gas mass, volume, and partial pressure in any node in POLCA-T. Update TR WCAP-16747-P to include this information.

(9) Please provide assessment information on the ability of POLCA-T to predict pressure drops for different flow rates, inlet subcoolings, powers, and different fuel bundle designs than in the test facility(ies) used for critical power testing.

(10) Please fully describe the nuclear instrumentation models in POLCA-T. These descriptions should include equations, references to other codes by name and version, and separate discussions for gamma and neutron sensitive instruments. Update TR WCAP-16747-P to include this information.

<u>RAI 6: Stability Evaluation</u>

The NRC staff has several questions regarding the stability evaluation. Please address the following items in regards to Appendix B of the submittal.

(1) Determination of a decay ratio (DR) requires accurate prediction of a steady state initial condition from which a perturbation allows a trace of transient dynamic behavior. The use of a stability methodology predicated on the DR must reliably predict steady state conditions for reactor operating states that are limiting from a reactor stability stand point. Provide statistically significant qualification of the nuclear design codes against data under limiting conditions of operation from a stability standpoint for high power density plants under single loop operation, recirculation pump trip, startup, and loss of feedwater heater conditions.

(2) Reactor dynamic behavior is a strong function of the void reactivity dynamic feedback. Provide an estimate of the sensitivity of the void reactivity coefficient bias, or uncertainty, to the exposure history in terms of spectrum hardness. Specifically, provide an estimate of the void reactivity coefficient bias and uncertainty predicated on nuclear parameter determinations under conditions typical of plants operating at originally licensed thermal power (OLTP) conditions and re-perform this estimation for conditions at high control fractions and void fractions. Based on this assessment update Appendix B of the TR to include provisions for use of the methodology for different operating domains or plant conditions.

(3) Describe the process for generating an appropriate nodalization for models used in stability analysis. Provide demonstration analyses that indicate the nodalization procedure, time step control, and numerical integration scheme do not adversely impact the numerical results of transient reactor behavior predictions. Provide a plot of the node-by-node Courant number for the hot channel for a plant included in the stability qualification based on the model used in the qualification.

(4) Describe how the results of the stability analyses are used in current operating BWR plants in terms of their implications to mitigating measures. For example, if a stability analysis is used to determine an exclusion zone, describe that process. If the stability analyses provide input to Detect and Suppress Solution set points, provide these details.

(5) Figure B.7-2 of TR WCAP-16747-P 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 TR.

(6) Explain the process for determining the nodal reactivity response to bypass or internal water channel voiding. If the effect of bypass or internal water channel voiding on nodal reactivity has been assessed, compare this effect to established

uncertainties in nodal parameters. Quantify expected bypass void fractions for transient conditions or expanded operating domains.

(7) Verify that POLCA-T can predict the onset of an instability by providing an analysis whereby increasing the reactor power or reactivity in a representative core model results in observed oscillations in the predicted transient traces.

(8) Provide additional details regarding the capabilities of POLCA-T to predict DRs for reactor cores that are highly stable (DR between 0.1 and 0.2).

(9) Qualify the thermal hydraulic model for the stability evaluation separately by providing an analysis of type one, type two, and loop oscillations for a non-nuclear experiment such as FRIGG.

(10) Based on FRIGG comparisons, the TR indicates that POLCA-T **[** **]**. The NRC staff does not agree that **[** **]** necessarily demonstrates conservatism. Density wave limit cycle oscillations for reactors are driven by the combined dynamics of the void reactivity, heat flux, and void propagation time through the core. Quantify based on available FRIGG data any uncertainty in the interfacial shear that may result in over- or under-predicting slip. Estimate the effect of this uncertainty on the magnitude of limit cycle oscillations based on representative void reactivity coefficients, fuel thermal time constants, and core flow rates.

(11) For each modern fuel design for which approval is sought for the stability methodology, provide the uncertainty for the local friction loss coefficients (i.e., for spacers and orifices). To the maximum extent possible, justify the values for these coefficients and their uncertainties using data.

(12) For each plant considered in the stability qualification, provide additional descriptive details of the operating conditions, namely the thermal power (absolute and fraction of OLTP), core flow (absolute and fraction of nominal), core size, and the radial and axial power shapes either measured by TIPs or predicted by the core monitoring system at the exposure point where the evaluations were performed.

(13) For limit cycle oscillations with high DRs, estimate the effects of potential feedback mechanisms from the balance of plant (i.e., feedwater heating) that are not explicitly modeled.

(14) For a sample of the cases considered in the qualification, perform a separate perturbation to initiate an oscillation. The results should be provided and the sensitivity of the DR to the perturbation technique should be quantified.

(15) Justify the extension of the DR acceptance criterion to plants with other vendors' fuel, expanded operating domains, or other core and plant design features that are not enveloped by qualification. If this is captured in the plant-specific design margin, provide in this TR the means for determining the plant-specific design margin.

(16) Typically, safety limit MCPRs are established to ensure that 99.9 percent of fuel rods avoid boiling transition. Explain how applying a prediction uncertainty of **[ ]** to the acceptance criterion adequately assures the same degree of protection against exceeding specified and acceptable fuel design limits (SAFDLs).

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

(18) Describe in greater detail the process by which the regional symmetry plane is determined.

(19) Verify the numerical solution efficacy for stability analyses by performing a simple analysis whereby **[ ]**. Provide the results of the analysis indicating that the numerical solution technique does not damp the oscillation at the outlet of the channel.

(20) Describe how the DR acceptance criterion provides adequate protection against exceeding SAFDLs when transient conditions may excite an instability at more limiting conditions than the steady state operating point, for example during a loss of feedwater event.

(21) Once POLCA-T has predicted the transient response to a perturbation, describe the procedure for determining the DR. Considering that a perturbation to the steady state system may excite numerous oscillatory modes at the beginning of the transient (for example multiple flux harmonics for a regional mode oscillation, type one channel oscillations during core wide oscillation perturbations, etc.); highlight any controls in this procedure that ensure the DR is representative of the oscillatory mode nearest to the limit cycle oscillation (or the most readily self-sustaining oscillation).

(22) Provide an assessment of POLCA-T and associated nuclear design codes to predict the radial power shape observed in the **[**
]. When performing plant- and cycle-specific evaluations for Option III plants, describe how the analysis is performed. Specifically address the determination of the symmetry plane for the oscillation.

(23) Please evaluate the effect of the update interval for the spatial solution of the kinetics solver on the MCPR for the range of transients being considered for POLCA-T. Does solving this equation for every amplitude/reactivity time step affect the results?

(24) Please evaluate the impact of using the implicit versus semi-implicit thermal hydraulic solver on the MCPR for the range of transients being considered for POLCA-T. What user guidance exists to help in choosing which solver to use and when? What are the effects of using the semi-implicit method in a component connected to a component using the implicit method?