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

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Responses to NRC RAI's Regarding

WCAP-16747-P Rev 0

RAI 1 Long Cycle Cores:

The staff has several questions regarding the application of POLCA-T to transients initiated from conditions typical of BWRs operating with long cycle durations (i.e. 24 month cycles). Please address the following items.

NRC RAI 1-1

Long duration cycle core designs substantial quantities of burnable poisons tend to be loaded in the fuel. In many cases these loadings may exceed 7 w/o Gd2O3 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.

Westinghouse Response to RAI 1-1

Modern, more aggressive, BWR operation strategies such as long cycles result in the extended use of burnable absorbers (BA), mainly Gd_2O_3. This concerns both the number of BA-rods per bundle as well as the BA contents in those rods. Well aware of this, Westinghouse has paid special attention to the proper modeling of modern bundle designs as well as to the validation of these models.

The specific use of high BA-content rods has been considered in PHOENIX/POLCA topical report, as reflected in the SER of CENPD-390-P-A, *"PHOENIX/POLCA are approved for analysis of ABB/CE fuel types up to and including 10x10 lattices with a maximum enrichment of 5 w/o UO_2. Non-ABB/CE fuel types may be analyzed assuming that analyses are performed consistent with (a) above. The code is approved for application to fuel with burnable absorbers composed of a mixture of UO_2 and Gd_2O_3 with concentrations up to 9 w/o Gd_2O_3. Application of the code to non-UO_2 fuel or the fuel using burnable poisons other than Gadolinia will need to be justified"*. The basis for this decision can be found, among others, in fig 5.12, page 103, of that report where a comparison against fuel rod gamma-scan measurements is shown. In that particular case, the bundle includes several measured [

]^{a,b,c} deviation at a highly challenging point in time (at end of first cycle, when the shelf-shielding weakens leading to a steep gradient in pin power over time).

An additional, highly relevant, source of information that helps to quantify the accuracy of the pin power evaluation in modern fuel is provided by a series of comparisons against gamma-scan measurements at individual rod level in well-characterized conditions. The most recent exercise involved comparisons against highly accurate measurements in the LWR-PROTEUS critical facility at Paul Scherer Institute (PSI) in Switzerland as part of a cooperation between Westinghouse, PSI and Swiss utilities. These experiments consisted in several measurements on a critical configuration of [ ]^{a,c}, with measurements of over [ ]^{a,c} in each case. Details of these critical experiments are provided in "Accurate Tools to Model Advanced SVEA Fuel Designs" (J. Casal et. al., Nuclear Technology, Vol. 151, July 2005, pages 51-59). From that reference, the following results (also presented at a meeting with the NRC staff in 2004, ref. LTR-NRC-04-70, Dec. 16, 2004) are shown. The deviations map presented here corresponds to an extreme case where this modern fuel design (with [ ]^{a,c} of the same kind as those loaded in a high power density GE-built BWR-6) is [ ]^{a,b,c} of the average power. [ ]^{a,b,c} as highlighted in the lower right corner of this figure. Moreover, several [ ]^{a,c} were measured (grey-marked), showing a prediction accuracy well within the accuracy of the [ ]^{a,c}

[ ]a, b, c

As described in the referenced document, within the LWR-PROTEUS phase 1 experiments, several critical configurations of relevance were measured including:

[ ]a,c

The following figure shows the overall statistics of these configurations for both total fission rate (representative of thermal power) and U-238 capture rate (representative of epithermal conditions) for more than 60 fuel rods belonging to the bundle in the central position. In most cases more than one enrichment/burnable absorber design were measured (configurations A/B) and even more than one control rod type (with either [ ]a,c

[ ]a,b,c

The [ ]a,b,c observed in the extreme cases of voided and controlled + voided conditions are mainly due to, as [

]a,c

In the same source ("Accurate tools…"), a brief description of another example of Westinghouse qualification effort can be found. In this case, it involves a comparison against a gamma-scan campaign at fuel rod level performed at a Swedish plant less than 3 months after the beginning of its last cycle of operation. These measurements included [ ]a,c As described there, no significant trends are observed in the [ ]a,c

An additional source of information is the comparison against higher order methods. As an example, comparisons of PHOENIX4 predictions for a SVEA96-Optima2 bundle against both the [ ]a,c and the [ ]a,c are presented here.

a, b, c

a, b, c

a, b, c

[a, b, c

a, b, c

Based on these and similar results Westinghouse concludes that all the evidence obtained so far, both from core follow and specific validation activities, support the assessment that the current, more advanced, fuel designs are handled by PHOENIX4/POLCA7 [ ]a,c compared to what was assessed in CENPD-390-P-A.

NRC RAI 1-2

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 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 TIP measurements for bundles where the power is suppressed for significant periods of operation (i.e. for leaking fuel bundles).

Westinghouse Response to RAI 1-2

Clarification: in addition to RAI 1-2, a number of RAIs (3-1, 3-3, 3-8, 3-9, among others) address, in different ways, NRC's concerns about the ability of Westinghouse nuclear design package (PHOENIX4/POLCA7) to model current bundle designs and operation conditions (long cycles, extended power uprates, etc). The text below is intended to both answer this specific question as well as to provide the basis to the answers to the other RAIs.

The ability to cope with different core conditions resides in the robustness and applicability of the methods implemented in the physics codes. Westinghouse aims to utilize methods that reliably perform over a wide range of conditions. For that purpose, ad-hoc solutions or bundle-specific, plant-specific or operation-specific assumptions are consistently avoided.

The qualification of Westinghouse nuclear design code package (PHOENIX4/POLCA7), in particular concerning the evaluation of safety-relevant parameters, described in CENPD-390-P-A, is based on a combination of analyses of core predictions against measurements, both from [ ]^{a,c} as well as [ ]^{a,c} comparisons. The extensive qualification effort during the development of these codes has been continued afterwards with additional core follow activities as well as additional gamma-scan campaigns. These activities lead us to conclude that the advanced models in these codes can accommodate the different core conditions resulting from e.g. long-cycle operation and/or power up-rates without any significant degradation of their predictive accuracy.

As an example, in addition to the two nodal/bundle gamma-scan campaigns reported in CENPD-390-P-A, Westinghouse described three additional campaigns at meetings with the NRC staff during 2005. The following is part of the material presented at those meetings.

Latest gamma-scan campaigns particularly interesting

[

]^{a,c}

[ ]a,c

[ ] a, b, c

[ ] a, b, c

[ ] a, b, c

a, b, c

[ ]

a, b, c

[ ]

As can be observed, albeit significant differences in core conditions at these two plants (power up-rates, cycle lengths, core compositions, etc) these validation exercises do not show any observable degradation of the nodal and bundle power predictions. Moreover, these observations are in [ ][a,c] At this point, it might be worth to point out that gamma-scan comparisons in most cases are only possible during planned outages, resulting in measurements of the conditions existing at EOC, so their results cannot be directly applied to the rest of the cycle.

[

]a,c The answer to RAI 1-4 discusses this subject in more detail.

An additional observation from these qualification exercises, and also one of the main drivers behind some of these measurements, is that [ ]a,c has been observed regarding the power distribution around the TIP strings which contribute to their signals (a concern raised in RAI 3-10 and 3-11). The reason for this is that PHOENIX/POLCA has clearly shown its ability to predict power levels in bundles over a wide range of exposures [ ]a,c Thus, each of the surrounding bundles being predicted with the same level of precision, [ ]a,c in the TIP comparisons. With this in mind, the fact that continued core follow activities (e.g. TIP comparisons) have shown that changes in core conditions [ ]a,c in predictive TIP-signal accuracy can reliably be extended to the nodal and bundle power predictions.

The use of TIP comparisons to evaluate the impact of the presence of control rods on the internal fuel rod power distributions is [ ]a,c the gamma (or neutron) flux generated by the four surrounding bundles to the TIP-string. Regarding the influence of control rods, the fuel rods most affected by the presence of a control rod are the ones most distant to the TIP-detector, [ ]a,c in the prediction of the internal pin power distribution. Therefore, a more valuable source of information is a gamma-scan comparison, at fuel rod level, of a controlled bundle. Such a comparison for a modern 10x10 bundle has been presented in the answer to RAI 1-1.

Additional information about the potential impact of the presence of control rods in long cycles on the accuracy in the prediction of the core power distribution as well as the ability of the TIP-system to detect those impacts are provided in the answer to RAI 3-11.

Concerning the so-called control blade history effect, a Control Rod History model is available in POLCA7. This model is utilized to [

]a,c as described as follows.

In the so-called control rod history branch cases a [

]a,c

[ ] a. c

where the [ ]a,c

The tables for the [

]a,c

[ ]a,c

where [

]a,c

[ ]a. c

[

]a,c

[ ]a,c

where [

]a,c

NRC RAI 1-3

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.

Westinghouse Response to RAI 1-3

Over the years Westinghouse has utilized gadolinia as a burnable absorber (BA) in standard fuel assemblies for the design of reload cores for both BWR and PWRs. Gadolinia is mixed homogeneously in UO2 with typical gadolinia contents in the range of 2 to 8 wt%. One of the impacts of introduction of Gd in UO2 on the fuel pellet characteristics is the reduction of the thermal conductivity of the pellet, due to the scattering of phonons with Gd atoms.

The thermal conductivity of UO2 fuel pellet consists of two contributions: conduction through lattice vibration (phonons), and conduction by electronic processes. The phonon component of the thermal conductivity may be written as:

[

]^{a,c} The impurity increases if gadolinia is present in the UO2 fuel pellet. Therefore for low gadolinia concentration, x (< 10 wt%), one may write:

[

]^{a,c}, as shown in equation A.1-6 of the Topical Report WCAP-15836-P-A, "Fuel Rod Design Methods for Boiling Water Reactor – Supplement 1".

The contributions of doped gadolinia on the thermal conductivity for different concentrations are shown in Fig. 1-3-1 below. The measured data from L.W. Newman, DOE/ET/34212-36 (BAW-1681-2) (1982) are also compared with calculation results. As can be seen, the calculated thermal conductivities match the measurement data satisfactorily for different levels of gadolinia concentrations without significant scattering.

The fuel thermal conductivity correlation used in Westinghouse codes, such an in STAV7 and POLCA-T is implemented in [

]^{a,c}

a, b, c

Figure 1-3-1. Comparison between calculated and measured thermal conductivity of (U, Gd)O2 as a function of temperature for different gadolinia concentrations.

NRC RAI 1-4

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

Westinghouse Response to RAI 1-4

The impact of different cycle lengths on the calculation efficacy of the code to predict reactivity levels can be shown by the following comparison of two different strategies. The cases selected include, in addition, a transition to higher power levels and a transition to different fuel designs (fuel with part-length rods (PLR) and/or from different vendors). In addition to the reactivity levels, information is provided about the accuracy in the power predictions (from TIP comparisons), to demonstrate that the behavior is [ ]^{a,c}

In the first case, the [ ]^{a,c}, which operates on [ ]^{a,c}, is shown over several cycles through its EPU.

[ ] a, b, c

a, b, c

[ ]

Both the reactivity levels (cold/hot cycle average keff) and the TIP comparisons (cycle average RMS errors) are [ ]^{a,c} It is important to point out that this particular plant is one of the [ ]^{a,c}

In the next case, the same information is provided in the following two charts regarding another high power density plant, [ ]^{a,c}, undergoing simultaneously [ ]^{a,c} This very significant change of core conditions over very few cycles is considered as one of the most challenging for the physics codes possible.

a, b, c

[ ]

a, b, c

[ ]

These plots show [ ]a,c behavior regarding the prediction of [ ]a,c over a number of cycles even under significant changes in the core. Based on the comparisons against cold critical measurements, [ ]a,c has been considered necessary for the calculation of [ ]a,c.

Responses to NRC RAI's Regarding

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RAI 2 Mixed cores

The 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.

NRC RAI 2-1

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, 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 (> 7 w/o Gd2O3 in several rods) under control states typical of operation for long cycle durations

Westinghouse Response to RAI 2-1

The drift flux correlation DF02 based on void measurement data from tests performed at Westinghouse's FRIGG Test Loop is used for modern fuel designs. The void data base includes void fractions for [ ]^{a,c} fuel, test series [ ]^{a,c} and for [ ]^{a,c} fuel designs with [ ]^{a,c}, test series [ ]^{a,c}.

Predicted versus measured void fractions calculated by POLCA-T for all simulated tests are plotted in the figure below.

[ ]^{a,b,c}

A summary of the statistics: number of test (N), mean average difference (m), and standard deviation (s) is shown below.

[ ] a, b, c

The following material provides a description of the experimental basis for the development of Westinghouse void correlation, as presented to the NRC staff in 2004 (ref. LTR-NRC-04-70, Dec. 16, 2004):

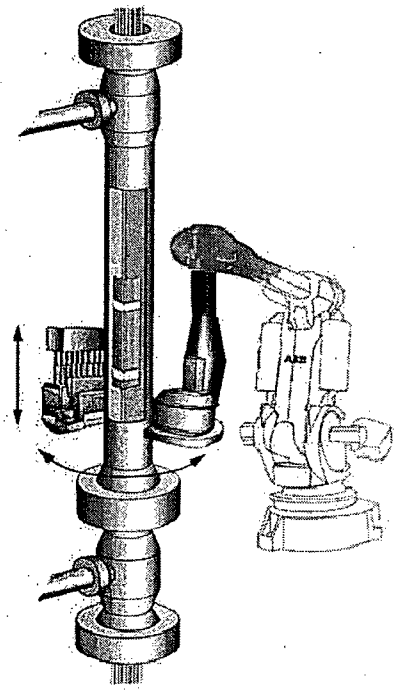
Void content predictions

The program included:

[

-] a,c

Void measurements at FRIGG (equipment)



a, b, c

[ ]

In conclusion, accurate measurements of realistic thermal-hydraulic conditions over a wide range have been performed at the Westinghouse FRIGG facility. These measurements confirmed [ ]^{a,c}. [ ]^{a,c} in the ability to predict void content [ ]^{a,c}. Moreover, as discussed in the answer to RAI 3-1, power uprate programs [ ]^{a,c} void levels than at pre-uprate conditions. Thus, the FRIGG-measurements data base [ ]^{a,c} for these applications.

See answer to RAI 1-2 for a discussion about the reactor physics predictions at high void.

NRC RAI 2-2

As discussed in the staff's SER for CENPD-390-P-A when applying PHOENIX/POLCA to transition cores, CENP 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.

Westinghouse Response to RAI 2-2

Westinghouse uses fuel specific data to model the thermal and hydraulic behavior of the non-ABB/CE fuel. This is accomplished by obtaining detailed pressure drop and flow split data for each legacy fuel design from the utility as part of the transition process. These data are typically obtained for several core power and flow combinations and a range of relative assembly powers. In conjunction with the detailed core heat balance data for the plant and the detailed mechanical design data for each legacy fuel design obtained from the utility, these data allow us to establish accurate thermal and hydraulic models in POLCA7 and other codes supporting the Westinghouse BWR reload licensing analyses. These data (i.e. pressure drops and flow splits) are physical quantities which allow us to tune the Westinghouse models in our codes to match these data without requiring any information regarding the Legacy fuel vendor's calculational models.

Uncertainties are typically required to assure that thermal limits are satisfied. Westinghouse has found that [

]^{a,c}. Therefore Westinghouse applies [

]^{a,c}. Consequently, the Legacy Fuel thermal limit which is calculated by Westinghouse [

]^{a,c}. This calculation is performed using the conservative Westinghouse methodology described in CENPD-300-P-A. As discussed in CENPD-300-P-A., the methodology used for establishing conservative OLMCPR's for Westinghouse fuel in conjunction with the hydraulic models established as discussed above, and a CPR correlation for the Legacy fuel established in accordance with CENPD-300-P-A, are used to evaluate the impact of potentially limiting AOO's (Anticipated Operational Occurrences) on the Legacy fuel. Then a bounding factor is applied to the resulting OLMCPR for the Legacy Fuel to assure it is conservative with 95% probability at the 95 % confidence limit. Consequently, the potential for increased uncertainties in the thermal analysis of the Legacy fuel is resolved by utilizing thermal limits for the Legacy fuel that are non cycle-dependent limits established by the Legacy Fuel vendor or conservatively bounding limits. Uncertainties associated with in-core monitoring of the fuel are depend on the Core Monitoring System, and Westinghouse uses sufficiently conservative uncertainties depending on the monitoring system and, in the case of Core Monitoring Systems not provided by Westinghouse, the information available.

NRC RAI 2-3

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

Westinghouse Response to RAI 2-3

Any CPR correlation to be incorporated in POLCA-T for any fuel design has to be reviewed by the NRC separately. WCAP-16081-P-A provides an example of NRC licensed CPR correlation for SVEA-96 Optima2 fuel. It means that for each fuel design for which approval is sought a CPR correlation will be submitted for approval. Chapter 12 "DROYUT and DNB CORRELATIONS" in WCAP-16747-P mentions in general that each licensed CPR correlation is linked to the POLCA-T code via a common library. **However, dry-out evaluations are not currently part of the scope of this particular Topical Report, thus no qualification efforts are pursued here.** A more comprehensive discussion about the usage of appropriate CPR correlations will be made in the coming Appendix C of WCAP-16747-P about Transients Applications.

NRC RAI 2-4

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

Westinghouse Response to RAI 2-4

The [ ][a,c] option in POLCA7 applies only to SVEA type fuel (four sub-bundles). [
][a,c] are modeled in POLCA7 for all fuel types that have such features.

NRC RAI 2-5

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.

Westinghouse Response to RAI 2-5

The pin power reconstruction methodology in POLCA7 is [ ]^{a,c} and applies [ ]^{a,c}. The only [ ]^{a,c} comes in the form of [ ]^{a,c} that are supplied as input to POLCA7. [ ]^{a,c} and, thus, they require no special treatment in POLCA7.

NRC RAI 2-6

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?

Westinghouse Response to RAI 2-6

The evaluation of transient responses are purely based on [ ]a,c In other words, the monitoring of the core with Westinghouse or any other core monitoring system does not impact the transient evaluations in any way. In the particular case of Westinghouse Core Monitoring System, the thermal margins evaluated are based on nodal power distributions corrected to accommodate for the differences [ ]a,c However, this [ ]a,c in any way the core parameters defining the initial conditions of the evaluated transients.

Responses to NRC RAI's Regarding

WCAP-16747-P Rev 0

RAI 3 Expanded Operating Domains

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

NRC RAI 3-1

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 (EPU). Modern core designs with higher numbers of higher power bundles at potentially higher bundle power to flow ratios warrants 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 SPU/ICF, EPU, MELLLA, 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.

Westinghouse Response to RAI 3-1

The answer to RAI 1-2 includes detailed information of the modeling of two different power uprate strategies (the so-called MELLLA and MELLLA+ conditions) at two of the BWR plants with the [

]^{a,c} A more detailed description of the "bounding" characteristics of the operating conditions of these plants can be found in the answer to RAI 11 to WCAP-15942-P-A. As observed there, all the available information, based on extensive core follow over many cycles before and after the power uprates, points clearly to an unchanged level of accuracy of the safety-relevant parameters.

The wide range of conditions covered by these plants and the limited practical impact of the power uprates on those conditions are illustrated in a series of 6 figures below. They show, for both plants

d) the bundle power/bundle flow conditions for all bundles at all calculated, during core follow, points during a whole cycle
e) the statistical distribution, for the same population (i.e. all channels, over the whole cycle), of the average channel void content
f) the statistical distribution, for the same population, of the outlet (i.e. maximum) channel void content.

a, b, c

a, b, c

a, b, c

[ ]

a, b, c

[ ]

a, b, c

a, b, c

As can be seen in these figures:

[

]a,c

Consequently, Westinghouse experience covers among the most challenging operating conditions at the present including significant power uprates at those plants. Moreover, those power uprates do not impact, as shown, significantly the range at which key parameters operate.

[ ]a,c is not individually considered in Westinghouse core analysis methodology. Westinghouse strategy, instead, is to evaluate [ ]a,c, which are directly affected by [ ]a,c. Nevertheless, its basic components [ ]a,c are evaluated as part of the qualification effort.

The accuracy of the void predictions was discussed in 2-1. The accuracy of the cell data as a function of void content has been the subject of several studies, among them the already mentioned LWR-PROTEUS experiment comparisons. Even comparisons against higher order methods have been performed. [

]a,b,c

a, b, c

[ ]

NRC RAI 3-2

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 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 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.

Westinghouse Response to RAI 3-2

As discussed in Section 5 of CENPD-300-P-A, an important feature of the Westinghouse methodology to assure that its reload fuel is compatible with the plant and Legacy fuel for a given reload application is to maintain [ ]^{a,c} within the same range as the original plant design or, if different, within the same range provided by the current Legacy fuel. This goal is achieved by [

]^{a,c}

These design features have enabled Westinghouse to [ ]^{a,c} in the interassembly by-pass and the SVEA fuel central channel and the water cross wings. As discussed in the Response to RAI 11 of WCAP-15942-P-A, the Westinghouse reload fuel experience has included two of the world's highest power density BWR plants: [
]^{a,c}

The applicability of nuclear instrumentation models to high power density applications is reflected by the [ ]^{a,c} in the response to RAI 1-4. The average power density [ ]^{a,c} during these measurements was [ ]^{a,c} which represents one of the highest power density BWRs in the world. The relative assembly nodal and assembly average standard deviations in the [ ]^{a,b,c}, respectively, clearly demonstrate the capability of POLCA7 to capture the LPRM/TIP response in a very high power density environment.

NRC RAI 3-3

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.

Westinghouse Response to RAI 3-3

As explained in the answer to RAI 3-1, the practical consequences of different core power levels on the conditions to be modeled are quite limited. A similar situation can be observed regarding long cycles. In the following figures the spectrum index (the thermal-to-epithermal flux ratio) in all the calculational nodes of two different reactors [ ]^{a,c} over a whole cycle are presented. The plots cover BOC, MOC and EOFP (end of full power) conditions and the histograms show the number of calculation nodes [ ]^{a,c} at the different spectrum index levels. As observed, [

]^{a,c} However, the extensive core follow and gamma-scan experience on a variety of reactors and cycle lengths, some of them presented in the answers to RAI 1-2, 1-4, and 3-1, does not [ ]^{a,c}. Moreover, such [ ]^{a,c} should not be expected since most of the nodes experiencing the hardest spectrum are found in the top of the core, due to the high void contents, regardless the length of the cycle. Consequently, frequently used spectral-shift operating strategies, where an initial top-peaked axial power profile is induced or reinforced in order to build up plutonium to be burned later on towards the end of the cycle, require trustworthy methods as well as for the reasons mentioned above. The accuracy estimates provided by TIP comparisons and gamma-scan comparisons [ ]^{a,c} these core conditions.

a, b, c

a, b, c

a, b, c

a, b, c

a, b, c

a, b, c

NRC RAI 3-4

Provide justification of the continued use of the pin and bundle power uncertainties provided in 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.

Westinghouse Response to RAI 3-4

The qualification basis for the pin power predictions in PHOENIX/POLCA are based on 2D comparisons against critical experiments (KRITZ reported in CENPD-390-P-A section 2.2, LWR-PROTEUS experiments in "Accurate Tools to Model Advanced SVEA Fuel Designs", J. Casal et. al., Nuclear Technology, Vol. 151, July 2005, pages 51-59) as well as gamma-scan comparisons at fuel rod level (8x8 and 10x10 bundles in CENPD-390-P-A section 5.3, additional measurements reported in the same paper mentioned above). These comparisons include highly actual bundles, as in the case of SVEA-96 Optima from operation in high density cores. All the information obtained so far clearly indicates that the new fuel designs are properly handled by PHOENIX/POLCA. The following summary was presented at the NRC on May 2005.

Power Uncertainties – PHOENIX4/POLCA7

[ ]a, b, c

[ ]a,c

NRC RAI 3-6

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 LHGR, and peak end-of-cycle nodal exposure. Provide quarter core map (assuming core symmetry) showing the bundle operating linear heat generation rate (i.e., maximum LHGR) and the 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.

Westinghouse Response to RAI 3-6

In the answer to RAI 3-1, as well as in 3-8, most of the information requested here has been provided for two plants prior and after a power uprate over several cycles.

Additionally, the following core maps, based on POLCA7 results, show typical BOC, MOC and EOFP (End of Full Power) conditions at [ ]^{a,c}. The MCPR results are limited to Westinghouse fuel bundles since the CPR correlations for non-Westinghouse bundles are not available.

a, b, c

a, b, c

[ ]

a, b, c

[ ]

a, b, c

[ ]

a, b, c

[

]

a, b, c

[ ]

NRC RAI 3-7

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.

Westinghouse Response to RAI 3-7

In the answers to RAI 3-1 and 3-3, a thorough description of the core conditions resulting of an EPU (MELLLA or MELLLA+) has been presented. As shown there, the differences in relevant parameters to prior-to-EPU conditions are quite limited and those conclusions are based on Westinghouse significant experience from both types of power uprates at two of the highest-density power plants in the world.

The use of core follow data to benchmark the analytical methods has not been modified in recent years, i.e. two different sets of data are required: a) input data describing the conditions of the plant at a given point and b) properly formatted (to meet the calculation code needs) data from the in-core instrumentation for the comparison against predicted signals.

Regarding the balance of plant data required as input to the codes, the main are the core thermal power, the core flow, inlet coolant conditions (temperature or enthalpy), control rod positions and time.

Regarding the in-core measurements required for validation of predictions, those normally used are provided by the traversing in-core probes (TIP) performed monthly for the calibration of the local power range monitors (LPRM). The TIP-detectors utilized for this purpose are either neutron-sensitive or gamma-sensitive. For validation purposes, gamma-sensitive detectors are preferred for being less sensitive to noise. These signals are normally [ ]^{a,c} by hardware/software associated to the core monitoring system.

The validation is centered on a comparison between gamma (or neutron) calculated TIP signals against the corresponding measured TIP signals (at LPRM calibration points) and normal statistics are employed, i.e. separate analysis of each TIP-channel as well as overall statistics: [

]^{a,c} In other words, no direct comparisons of results from the core monitoring system against off-line measurements are considered for the validation of the analytical methods.

As an example of the ability of PHOENIX4/POLCA7 to predict the power distribution on a high power density plant, nodal and radial (bundle) TIP comparisons are provided for two consecutive cycles [ ][a,c] in the figure below.

a, b, c

[ ]

NRC RAI 3-8

The objective of this review 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.

Westinghouse Response to RAI 3-8

The following tables summarize the most relevant core follow data over several cycles for the [ ]^{a,c}, operating, as already mentioned, at the highest power density in the world. These cycles start at the transition to the new power level (MELLLA+). As already discussed in 3-1, [ ]^{a,c}

[ ]^{a,c}

[ ]^{a,b,c}

[ ]a,c

a, b, c

[ ]a,c

a, b, c

[ ]a,c

a, b, c

NRC RAI 3-9

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.

Westinghouse Response to RAI 3-9

The operation conditions of plants undergoing EPU programs (both of MELLLA and MELLLA+ type) have been illustrated in the answer to RAI 3-1. There, all the calculation points from core follow over a whole cycle, i.e. all the conditions those cores experience during its operation, were considered when gathering the data presented there. As shown there, the range of conditions or parameters does not deviate significantly from the conditions or parameter values found prior to the power uprate.

As discussed in the answer to RAI 1-1, both TIP comparisons and gamma-scan comparisons support the conclusion that no significant changes have been observed in the prediction capabilities of Westinghouse physics codes. It is worth to note that in particular the TIP measurements are performed (and compared) over the whole cycle, which means that they cover the whole core flow range utilized during operation. As an additional example, the following table summarizes a number of TIP comparisons performed during the start-up, over several cycles, at a [ ]a,c operating after a [ ]a,c For each cycle, two different conditions are shown: a TIP measurement performed at off-rated conditions (low power/low flow) together with the first TIP measurement at rated conditions. As shown there, the resulting RMS-errors are well within the levels reported in CENPD-390-P-A.

[ ] a, b, c

Regarding the specific accuracy of POLCA-T at off-rated conditions, beyond the conditions presented here above, the lack of extensive measurements at such conditions is somehow compensated by the fact that the impact of potential inaccuracies in those predictions on the relevant-to-safety results of POLCA-T are already factored in the validation exercises against plant measurements included in the qualification suite. Examples of this are the comparisons against core stability measurements or recorded transient events. In other words, the resulting accuracy of those predictions already account for the accuracy of the modeling of the initial conditions.

NRC RAI 3-10

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.

Westinghouse Response to RAI 3-10

The intrinsic limitations of the TIP measurements have been a matter of specific attention at Westinghouse and its Customers. This has been one of the key motivations behind the decision of performing extensive whole-bundle [ ]^{a,c} measurements at their plants. These measurements, which provide reliable information of the power conditions over the previous few weeks of operation, both quantified the accuracy of the nodal and bundle power predictions as well as confirmed the suitability of the TIP-measurements as a source of data describing the power distribution in the core, both before and after the power uprates. This is evident, for example, in the predictions at [ ]^{a,c} (see answers to RAI 1-2 and 3-8), where the [

]^{a,c} In this way, the information collected becomes insensitive to the specific loading pattern.

During those measurement campaigns, specific attention was paid to scan the four bundles surrounding some of the TIP-positions as in the example shown below where the differences between calculated and measured gamma scans over the four bundles around the TIP were evaluated.

Nevertheless, the most reliable and comprehensive source of data for the quantification of the accuracy of the power predictions obtained with PHOENIX4/POLCA7 are the gamma-scan measurements presented in the answer to RAI 1-2, which are not affected by the intrinsic limitations of the TIP-measurements. The availability of these, highly relevant measurements, render less critical the use of TIP-comparisons for validation purposes.

a, b, c

[ ]

As observed in this case, the [ ]a,c.

Additional information about the potential masking of power sharing errors on TIP signals is provided in the answer to RAI 3-11.

NRC RAI 3-11

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.

Westinghouse Response to RAI 3-11

As part of the answers to RAIs 1-2, 1-4, 3-4 and 3-6, the issue of the code accuracy at more challenging operation strategies has been discussed to some extent. Regarding the specific aspect of the [ ]^{a,c} of local differences (axial and nodal) during the normalization and combination of data during the TIP comparisons, gamma-scan comparisons have added highly relevant information by [ ]^{a,c}, as already mentioned in the answer to RAI 3-11.

Additionally, the following example is provided here. In this case, TIP comparisons from a TIP-location clearly affected by the presence of different control rods at different times over a cycle at [ ]^{a,c} are presented. In this case, with the plant already at the highest power level [ )]^{a,c}, the measured (TIPMEA) and calculated (TIPGAM) nodal values are presented at several points over the cycle. At each point, the core conditions, the location of the neighboring control rods around the TIP trace and the accuracy of the TIP predictions (expressed as the standard deviation of the relative errors, normalized over that particular TIP trace at that point in the cycle) are provided. The figures show the ability of the code package to predict quite different conditions with a very consistent accuracy level. The values obtained are fully in line with the values reported in CENPD-390-P-A.

a, b, c

[ ]

At this point it is worth to note that any significant [ ]a,c distribution ought to have resulted in an [ ]a,c when data are analyzed in this way. This is due to the fact that the conditions evaluated here cover a wide range of controlled/uncontrolled configurations over time and significant changes in axial power distribution are obtained due to both the depletion of the core and those control rod movements. This should have resulted in [

]a,c, which would have made these comparisons [ ]a,c with each other. Nevertheless, these statements are not as conclusive as the evidence provided by the gamma-

scan comparisons previously discussed, which have clearly indicated [
]a,c

Responses to NRC RAI's Regarding

WCAP-16747-P Rev 0

RAI 4 Code Legacy

The 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 review. Please address the following items.

NRC RAI 4- 2

List all modeling assumptions in the development of the formulae in PHOENIX and POLCA that were not explicitly stated in CENPD-390-P-A. If code-to-code comparisons were used to normalize or adjust models in an empirical manner, provide the details.

Westinghouse Response to RAI 4-2

No empirical adjustments to models were made and no essential assumptions other than those listed in CENPD-390 are relevant.

NRC RAI 4- 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 topical report WCAP-16747-P, otherwise provide the guidance as stated in internal procedures or manuals for determining the base and branch cases in the update.

Westinghouse Response to RAI 4-3

In order to support the cross section (XS) data generation for the POLCA7 cross section model given by eq. (4.1) in CENPD-390-P-A, a calculation matrix of the kind supplied here below is applied in the lattice calculations. In Westinghouse methodology there is [ ]^{a,c}, which are periodically revised to guarantee the adequate modeling of different applications. An example of such revision is the addition of [ ]^{a,c} for an improved modeling of isothermal temperature coefficients. Any revision of the values recommended for the different state parameters in the cross section model of POLCA-7 is properly qualified (verified and validated) and documented prior its implementation for production applications. The new recommended [ ]^{a,c} is communicated to the code users according to current practices, i.e. by a revision of the Reload Design Procedure.

[

]^{a,c}

All branch cases with

]^{a,c} are performed for all instantaneous coolant conditions (i.e. the dependence on coolant density is built [ ]^{a,c}) whereas [ ]^{a,c} cases are only performed for HFP sub-cooled and HZP - CC conditions.

Coolant density history conditions: *The HFP coolant conditions specified above.*

Instantaneous coolant density branch cases, [ ]^{a,c}, as well as the one for [ ]^{a,c} are performed for all [ ]^{a,c} whereas all other instantaneous branch cases are performed for the reference [ ]^{a,c} only.

Of all these parameters, the one leading to extrapolations is [ ]^{a,c} Regardless of the fact that any potential impact on this extrapolation is already factored in the comparisons utilized to characterize the accuracy of Westinghouse code package [ ]^{a,c}, this has been the matter of special attention as reported to NRC during the 2005 meeting. Among the information presented at that meeting was an exercise presented here below, where a single channel at conditions representative for the average and hot channel in a high density power plant was analyzed. For this purpose, cross sections generated with PHOENIX at the standard [ ]^{a,c} and equivalent cross sections generated with a [ ]^{a,c} at the exact [ ]^{a,c} of each calculational node were utilized in a 1-D version of POLCA without thermal-hydraulic feedback [ ]^{a,c}. This comparison demonstrated that both the interpolation and extrapolation of [ ]^{a,c} performed was [ ]^{a,c} assumed in CENPD-390-P-A.

Verification – 1D Comparisons

[

–]^{a,c}

a, b, c

[ ]

a, b, c

[

]

Verification – Summary

[

]a,c

NRC RAI 4- 4

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

Westinghouse Response to RAI 4-4

The POLCA-T code supports the steady-state, transient, and accident analyses comprising the Westinghouse reload fuel methodology. The Westinghouse reload fuel methodology either utilizes well characterized uncertainties or clearly conservative assumptions which bound the application of realistic uncertainties. The general treatment of the various analyses with regard to the use of conservative assumptions or the use of uncertainties is discussed in CENPD-300-P-A. The magnitudes of specific uncertainties are addressed in the methodology documents describing the specific applications.

[ ]a,c are specific to the PHOENIX4/POLCA7 code system and are described in CENPD-390-P-A. Further justification and the continued applicability of the magnitudes of these uncertainties are discussed in items 3-3 and 3-4 of this document. The [ ]a,c is used to establish [ ]a,c. [ ]a,c are used to support the establishment of relative assembly power and flow uncertainties to be used for applications of the Westinghouse Plant Core Monitoring systems.

In addition, while not used directly in the establishing of the [ ]a,c using the Westinghouse methodology described in WCAP-15942-P-A, relative uncertainties in [ ]a,c established for the PHOENIX4/POLCA7 system of codes were used in the evaluation of data used to support the fuel rod performance analysis methodology development described in WCAP-15836-P-A.

Responses to NRC Requests for Additional Information are included in the release of the approved version of the topical in question.

NRC RAI 4- 5

Explain how a bundle specific R-factor is determined.

Westinghouse Response to RAI 4-5

The bundle R-factor distribution is provided by POLCA7 to establish CPR using appropriate licensed CPR correlation. This [ ]^{a,c} for the entire core is used in POLCA-T for calculating the CPR values during transients. The R-factor is a part of a specific CPR correlation and is licensed together with the correlation (see also the answer to RAI 2-3).

NRC RAI 4-6

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?

Westinghouse Response to RAI 4-6

The direct energy deposited into the moderator is a fraction of the prompt fission power in the active coolant and in the bypass (external and internal). This fraction of the power is calculated using a user input and consists of a [ ]^{a,c} The direct heat dependency on density is derived from the lattice code calculations. Thus the density, [ ]^{a,c} is accounted for during the transient simulation. The code assumes the [ ]^{a,c} on density for both bypass flows, external and internal.

NRC RAI 4- 7

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

Westinghouse Response to RAI 4-7

The pump curves are user input and can be given in two different ways: Homologous curves for each unique pump in the model describing the pump head and hydraulic torque as a function of pump speed, volumetric flow rate and void fraction. This pump description is made up of four ratios for head, flow, speed and torque for each quadrant with its reference values. The second way to specify a pump curve is to provide performance input data for pump head and hydraulic torque as a function of flow rate for the first quadrant for each pump.

The pump performance data is normally based on data provided by the pump manufactures or in some cases based on additional test data for some Westinghouse BWR plants.

NRC RAI 4- 8

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

Westinghouse Response to RAI 4-8

The time step size algorithm takes into account:

- Sets upper and lower limit of time step which can be altered during the simulation by the user defined table.
- Time step allowed by the shape of the disturbance, so that at least each knot in the disturbance is simulated.
- Time step allowed by the neutron kinetics. The kinetic time step limitations are based on allowed neutron flux change over a time step, both a maximum and minimum relative change during one time step.
- Time step allowed by hydraulics. The hydraulic time step is determined from the convergence criteria for the state variables, pressure, void, liquid and gas temperatures, liquid and gas velocities, boron concentration and partial pressure of non-condensable gases.
- In addition to the above time step criteria a Courant number step criterion can be opted, where a selected Courant number is used to determine a candidate for time step.

The used time step is [ ]^{a,c}

NRC RAI 4-9

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?

Westinghouse Response to RAI 4-9

As it was mentioned in the answer to the RAI 4-8 the neutron kinetics time step limitations are based on allowed neutron flux change over a time step, both a maximum and minimum relative change during one time step. In our methodology for RIA we often use [ ]^{a,c} that leads to an almost [ ]^{a,c} Due to the very high neutron flux changes that take place in the RIA the upper limitation of the time step is set normally to [ ]^{a,c} For a given set of analysis the user is also required to investigate the time step effect on the peak power value, [ ]^{a,c} The time step limit obtained in such a sensitivity study is used in further analyses.

Responses to NRC RAI's Regarding

WCAP-16747-P Rev 0

RAI 5 Individual Models and Separate Effects Qualification

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

NRC RAI 5- 2

Verify whether PHOENIX is exercised with a 34 group or 89 group neutron cross section library and the 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.

Westinghouse Response to RAI 5-2

The PHOENIX library used in production is the [ ]^{a,c} generated and validated as described in CENPD-390-P-A. The [ ]^{a,c} has been based on observed biases in the so-called TRX criticals and the correction introduced is aimed to reduce the biases on these experiments while keeping a good agreement with the more relevant [ ]^{a,c} which cover a broader and more relevant range. This has been reported in *B. Fredin et. al. "Processing and Application of ENDF/B-VI in LWRs: Critical Experiments", Trans. Am. Nucl Soc., Vol 73, p. 419, 1995.* Thus, this correction is considered relevant to LWR-systems in general and not vendor specific bundles. The extensive validation effort discussed in answer to RAI 1-1 on modern fuel designs has confirmed that the accuracy obtained with that library is fully in line with the accuracy estimates provided in CENPD-390-P-A.

NRC RAI 5-3

If approval is sought for application of POLCA-T to 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.

Westinghouse Response to RAI 5-3

[ ]a,c PHOENIX4/POLCA7 nor POLCA-T is intended to be used for [ ]a,c at this time. Thus, [ ]a,c has been pursued to demonstrate the abilities of these codes for that kind of applications.

NRC RAI 5- 4

Please evaluate the sensitivity of burnup predictions on minimum critical power ratio (MCPR) for the transients of interest.

Westinghouse Response to RAI 5-4

This topic is not currently part of the scope of this particular Topical Report, thus no qualification efforts are pursued here. Required burnup determination frequency to establish representative isotope, power and flow distributions during an operating cycle is part of the transient methodology and will be treated in Appendix C of WCAP-16747-P about Transients Applications.

NRC RAI 5-5

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

Westinghouse Response to RAI 5-5

This topic is not currently part of the scope of this particular Topical Report, thus no qualification efforts are pursued here. The evaluation of the effects of power distribution uncertainty on MCPR for the transients of interest is part of the transient methodology and will be treated in Appendix C of WCAP-16747-P about Transients Applications.

NRC RAI 5-6

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

Westinghouse Response to RAI 5-6

Gamma-smearing of pin powers is done in the [ ]^{a,c} in three basic steps. First, the gamma energy deposition rates in the fuel pellet and cladding [ ]^{a,c} and the ratio of [ ]^{a,c} produced by the lattice code to obtain ['gamma-deficient' pin powers.]^{a,c} In the second step, the pin-wise gamma energy deposition rates in the fuel pellet and cladding are [ ]^{a,c} to obtain [ ]^{a,c} In the final step the [ ]^{a,c} are normalized such that they add up to the total number of fuel pins. This yields the [ ]^{a,c} that are passed on to POLCA7.

NRC RAI 5-7

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?

Westinghouse Response to RAI 5-7

See answers to RAIs 1-1 and 2-1 for a discussion about the modeling accuracy of voided conditions.

As discussed there, the individual contribution of the accuracy of the void reactivity coefficient is [ ]^{a,c} The resulting uncertainty in (nodal, bundle, rod) power predictions are [ ]^{a,c}

NRC RAI 5-8

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

Westinghouse Response to RAI 5-8

The partial pressure of non-condensable gases (called nc) (p_{nc}) is one of the eight state variables in POLCA-T for a volume cell as stated in WCAP-16747-P. [

]^{a,c}

The conservation of non-condensable gases is controlled by a balance equation in each volume cell for the mass of non-condensables (m_{nc}).

[ ]^{a,c}:, where w_{liq} , w_{gas}

are the mass flow rates for each phase, C_{nc} is the fraction of non-condensable species in the actual phase in the donor cell and Γ_i is a source/sink term to take into account production or consumption of non-condensable due to chemical reactions. The sum goes over all flow paths directed into the volume cell and all flow paths directed out of the volume cell.

[

]^{a,c}

[

]^{a,c}

The mass flow rates of liquid and gas in one flow path are

$$w_{liq} = u_{liq} \cdot \rho_{liq} \cdot A \cdot (1-\alpha)$$

$w_{gas} = u_{gas} \cdot \rho_{gas} \cdot A \cdot \alpha$

The densities and void fractions are taken from the donor cell.

Where

w	is the mass flow rate
u	is the velocity of the phase
A	is flow area
α	is the void content
ρ	is the density

The conserved quantity i.e the mass of non-condensable (m_{nc}) is a function of volume, void, phase densities and concentrations of non-condensable

[ ]a,c

here:

$\rho_{gas} = \rho_{ncg} + \rho_{steam}$

$$\rho_{liq} = \rho_{water}\left(1 - x_{ncl} + x_{ncl}\frac{M_{nc}}{M_{water}}\right)$$

with

$\rho_{water} = \rho_{water}(T_{liq}, p)$ from steam tables

The liquid density is based on the assumption that each molecule of non-condensable occupies the same space as one molecule of water which is reasonable considering the low concentrations of dissolved non-condensable that can occur.

[

]a,c The equation for each volume cell is entered into the complete set of thermo-hydraulic equations that are solved and integrated in time using the numerical method described in WCAP 16747-P.

NRC RAI 5-10

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

Westinghouse Response to RAI 5-10

POLCA-T makes use of TIP and LPRM detector models of the licensed POLCA7. Models for both gamma and neutron sensitive instruments are described in the approved topical report CENPD-390-P-A "The Advanced PHOENIX and POLCA Codes for Nuclear Design of Boiling Water Reactors".

Responses to NRC RAI's Regarding

WCAP-16747-P Rev 0

RAI 6 Stability Evaluation

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

NRC RAI 6-1

Determination of a decay ratio 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 decay ratio 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 SLO, RPT, startup, and LOFWH conditions.

Westinghouse Response to RAI 6-1

The qualification of Westinghouse nuclear design codes, PHOENIX4/POLCA7, is extensively addressed in CENPD-390-P-A. The validation covers normal operating conditions including low flows on reactor power-flow maps, which is of interest in stability analyses. Moreover, as described in the answer to question 1-2, a continued qualification effort after the submittal of CENPD-390-P-A based on gamma-scan comparisons of modern fuel bundles has confirmed the reliability of these tools. The specific quantification of the accuracy of the stability predictions, based on the initial conditions calculated with these tools, is also performed by comparisons against a wide range of measurements at realistic core conditions. Consequently, any impact of the steady state initial conditions on the stability predictions is accounted for in those comparisons.

NRC RAI 6-2

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 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 topical report to include provisions for use of the methodology for different operating domains or plant conditions.

Westinghouse Response to RAI 6-2

The accuracy and reliability of the void and void reactivity coefficient predictions are discussed in more detail in the answers to questions 2-1 and 3-1. As explained there, in Westinghouse methodology the accuracy of individual models [ ]^{a,c} On the other hand, [ ]^{a,c} under realistic, relevant conditions are utilized to quantify the [ ]^{a,c} utilized in the code package under qualification. Regarding the specific case of interest here, i.e. stability predictions, Westinghouse has access to a wide data base of stability measurements in Europe, covering real plants of different design from different plant vendors loaded with fuel of different design in cores with different mix levels. Therefore, the validation of POLCA-T for stability applications is mainly made on [

]^{a,c}

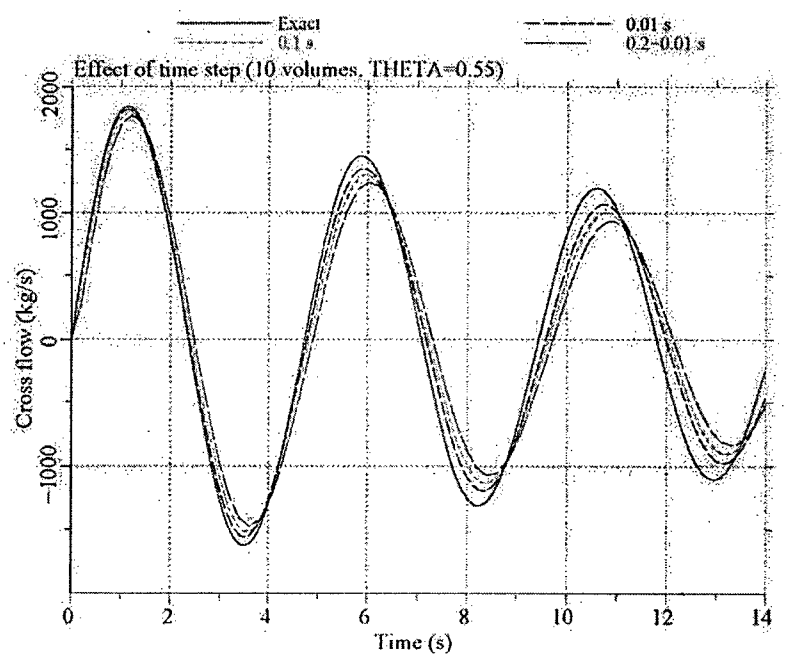
NRC RAI 6-3

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.

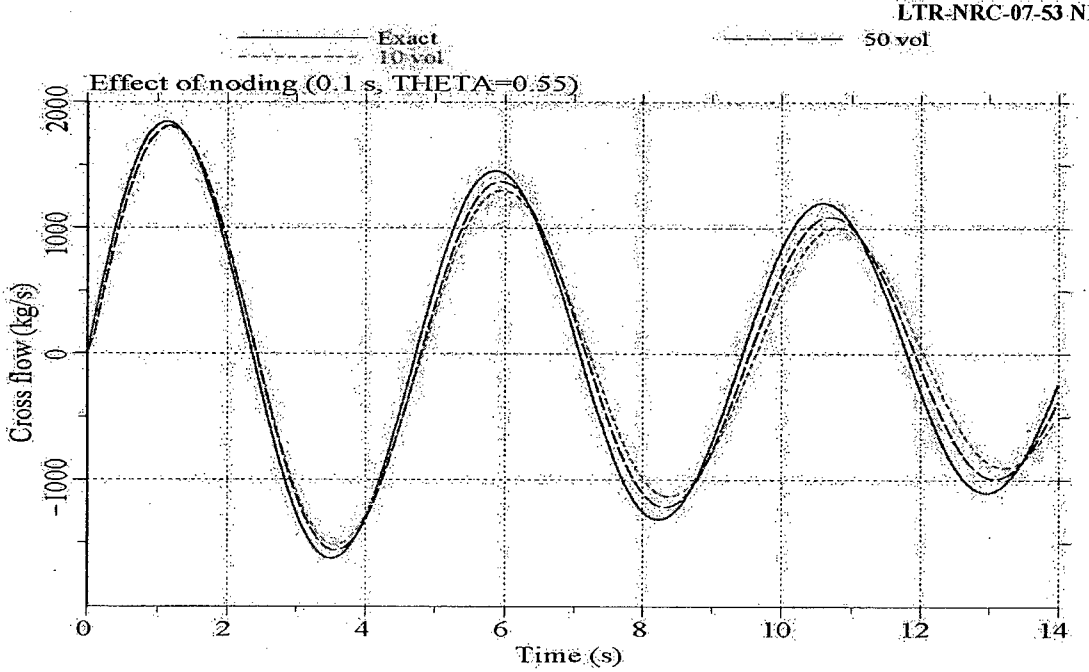
Westinghouse Response to RAI 6-3

The time integration method and its impact on the prediction of the hydraulic stability are discussed in a paper presented at ICONE 15. (ICONE15-10033). In the paper results for different time integration methods, different time steps and different number of volume cells are compared. As reference an "exact" numerical solution of an oscillating liquid in a U-tube is used. The outcome of this study is that semi implicit time integration method is a must, the time step and number of volume cells has not so big impact on the result, if it is a moderate change. A moderate change is considered up to a factor of five in the number of cells and a factor of ten in the number of time steps.

The figures below show the effect of time step and effect of number of volume cells. In the responses to the RAIs 6-19 and 6-24 the effect of numerical time integration method will be demonstrated.



Effect of time step choice for 10 cells model with $\Theta = 0.55$



Effect of number of cells for equal time step and integration parameter,Θ

For core calculations the core model is normally inherited from a POLCA7 core model. The axial nodalization in POLCA7 is arbitrary and is user specified. Normally the fuel assembly is divided [ ]^{a,c} For the neutronics calculation it is desirable that the nodes are [ ]^{a,c} The TH-nodalization (volume cells) of the core in POLCA-T is normally taken to be [ ]^{a,c} For the nodalization of the reactor pressure vessel outside the core the length of the nodes should be of the same order of magnitude as in the core.

NRC RAI 6-6

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.

Westinghouse Response to RAI 6-6

Nodal reactivity response to bypass or internal water channel voiding is handled by [ ]^{a,c} The user has the opportunity to specify in the input data whether to account for [ ]^{a,c} or not. The input is given for each fuel assembly type separately.

POLCA-T uses the POLCA7 model to account for bypass density changes. The code calculates the effective nodal moderator density ρ_{cool}^{eff} as a function of [

]^{a,c}

where C_{0i}, C_{1i}, and C_{2i} are density coefficients derived from the lattice code calculations,

A_{cool} and A_{bpi} are respective flow areas of the moderator in the bundle and in the bypasses,

$\rho_{bp}^{lattice}$ is bypass coolant density assumed in the lattice code cross section generation and i is an index running over the bypass 'channels'.

The density coefficients above must be computed with a lattice code [ ]^{a,c} Alternatively, a widely used model based on [ ]^{a,c} can be used by specifying C_{1i}, and C_{2i} above as [ ]^{a,c}, and C_{0i} as [ ]^{a,c} This gives

[

]^{a,c}

If all density coefficients ($C_{1i/2i/3i}$) are set to zero the density changes in the bypass during the transient are not accounted for which is a conservative assumption in the transient simulations.

NRC RAI 6-7

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.

Westinghouse Response to RAI 6-7

A demonstration to verify the capability has been performed for [

]a,b,c

The reactor power responses in these cases are shown in the following plot.

[ ]a, b, c

NRC RAI 6-8

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

Westinghouse Response to RAI 6-8

The DR regime below [ ]^{a,c} is in general difficult to evaluate for several reasons. The most significant is that the measurements uncertainties are of the same magnitude as the values themselves. Also, it is very difficult to deduce a DR from the power trace since the oscillations dies out very fast. An additional issue is that the initial perturbation influence may become significant on the deduced DR.

The hypothetical test case [ ]^{a,c} shows that nothing is 'destroyed' be the perturbation. The conclusion based of such a test is that the code predicts [ ]^{a,c} correctly, and must therefore perform correctly on the interval [ ]^{a,c} (the physics processes are the same and no discontinuities are expected). However, the uncertainty on the interval can not accurately be quantified. Therefore, attempts to accurately calculate DR in the interval [ ]^{a,c} is not performed since the accuracy is not possible to verify. Also, from the safety point of view such low DR's are of no interest to analyze independently of the accuracy of the calculation.

NRC RAI 6-11

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.

Westinghouse Response to RAI 6-11

The approval of each fuel design is subject for a fuel specific topical report. Uncertainties related to the fuel specific data are verified and documented in that process. Data for fuel from other vendors[2], including uncertainty of these data, must be supplied by the utility. The local friction loss and flow restriction coefficients are based on pressure drop measurements for the components in each fuel type and cover the conditions for stability analysis.

To demonstrate the impact of the uncertainties two runs based on the [ ]^{a,c} c19a case 3 have been performed. The applied uncertainty of [ ]^{a,c} is larger than the ones obtained from the pressure drop measurements.

[

]^{a,c}

Both studied cases have uncertainties larger than the evaluated values from pressure drop measurements. The DR calculation validation against plant measurements includes the uncertainties in the restriction and friction coefficients. Thereby, the method uncertainty and the design margin cover the uncertainties in the pressure loss coefficients.

NRC RAI 6-12

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.

Westinghouse Response to RAI 6-12

The absolute power, the nodal (F_nod), axial (F_ax)) and radial (F_rad) peaking factors are given in the table below. So is the relative core flow for the [ ]a,c. For the [ ]a,c reactors nominal core flow is not defined, and a relative core flow can therefore not be calculated. The peaking factors are obtained from the steady state calculations with POLCA-T.

a, b, c

[ ]

a, b, c

[ ]

[ ] a, b, c

The core size in terms of number of assemblies is already specified in the topical report. All three reactors are built with a 30 by 30 grid.

NRC RAI 6-14

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 decay ratio to the perturbation technique should be quantified.

Westinghouse Response to RAI 6-14

A demonstration study using different types of perturbations has been performed. The [ ]a,c c19a case 3 measurement has been used as a reference (case 0 below) for the tests. The following perturbations were performed;
[

]a,c

[ ]a, c

The spread in DR is [ ]a,c (standard deviation) and in frequency [ ]a,c Hz. It is worth noticing that even the cases that can be expected to be most challenging, [ ]a,c, do not deviate significantly from the rest of the cases. The conclusion is that there is no dependency on the perturbation technique as long as it is reasonable.

NRC RAI 6-16

Typically, safety limit minimum critical power ratios are established to ensure that 99.9% 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 SAFDLs.

Westinghouse Response to RAI 6-16

The stability methods and methodology presented in this Topical report covers the basic stability evaluation, *i e* calculation of decay ratio and resonance frequency at different operating conditions. That is, decay ratios below and up to the stability limit. Under these conditions any disturbances will 'die out' and the SAFDLs are fulfilled as long as the static SAFDLs are fulfilled. High amplitude power oscillations on the other hand, limit cycle or increasing amplitudes, could cause dry out. The DIVOM methodology is applied for such scenarios to calculate set points to the detect and suppress algorithms.

NRC RAI 6-17

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

Westinghouse Response to RAI 6-17

The fundamental physics (the TH-neutronics-TM coupling) on the local (nodal) level is the same independent of type of oscillation and the code can therefore be expected to be able to handle all oscillation modes, with the same accuracy, if it can handle one of them. It is therefore feasible to apply the same uncertainty and acceptance criteria for all oscillation modes. The number of recorded measurements (and incidents) of regional and channel oscillations is very limited. A statistical analysis for other modes than the global mode is therefore not feasible