Appendix A: Review of Westinghouse Responses to NRC staff RAIs

RAI 1: Long-Cycle Cores

RAIs 1-1, 1-2, and 1-4

The NRC staff has reviewed the information contained in the responses to NRC staff RAI 1. Westinghouse provided data to support the calculational efficacy of the PHOENIX4/POLCA7 code system to model those neutronic aspects of long-cycle cores. The data consists of:

- Experimental results of gamma scan tests performed at the LWR-PROTEUS critical facility at the Paul Scherer Institute in Switzerland,
- Gamma scan campaigns at **[ ]**
- Sophisticated transport code comparisons, and
- Core follow data at **[ ]**

The experimental results compared **[ ]** gamma scanned pin powers for a **[ ]** fuel bundles. The critical assembly also included **[ ]** The response provides specific details for one particular gamma scan of a bundle under controlled conditions. The results indicate good agreement between the neutronic model and the critical experiment with an error in predicted pin power **[ ]** that was controlled. For the voided conditions, the **[ ]** However, as stated in the response, the **[ ]** The results also indicate robust performance of the PHOENIX4 methods **[ ]**

The gamma scan campaigns at **[ ]** The **[ ]** plant. The cycle considered as part of the qualification data is the cycle **[ ]** core. The **[ ]** Results of the gamma scan campaign indicate very good agreement in pin power distribution, axial power distribution, and bundle power distribution. The root-mean-square errors for the **[ ]** core nodal and bundle power distribution are **[ ]** respectively. These are within the established uncertainties for the PHOENIX4/POLCA7 codes as referenced in CENPD-390-P-A (see page 77 of the TR).

The additional gamma scan information provided indicates that the predictive accuracy of the PHOENIX4/POLCA7 codes does not degrade with respect to bundle geometry or high bundle power. The accuracy of modeling the power distribution for cores operating with modern spectral shift strategies requires a robust technique for modeling the control blade history effects. Westinghouse supplied the details of the weighting functions used to determine

nodal parameters based on lattice depletion branch cases. The combination of the controlled and uncontrolled depletion cases, associated branches, and the build-up and decay phase weighting functions provide the means for tracking the control blade history effect in the PHOENIX4/POLCA7 code system. The NRC staff reviewed the formulation of the control blade history model and found that the theoretical representation is consistent with industry practice, and the accuracy is adequately demonstrated through the gamma scan results.

Therefore, the NRC staff finds that the gamma scan campaigns provide adequate reasonable assurance that the code system is sufficiently sophisticated to model those neutronic phenomena important to the determination of power distributions in long-cycle BWR cores.

The code-to-code comparisons were carried out using the HELIOS (higher order method) and MCNP (Monte Carlo transport) codes. The comparisons indicate agreement within the established error in pin powers. Cases with high radial power peaking and void fractions were considered as part of the benchmarking. The NRC staff finds that this confirms the solution technique for the PHOENIX4 lattice code for modern fuel designs and operating strategies.

The qualification information provided in the response to RAI-1 also addresses the eigenvalue determination. The response considered cycle tracking data at **[**
] In these **[**

] The eigenvalue predictions and critical configurations indicate that there is no significant change in the POLCA7 predictive capabilities with (1) cycle length, (2) batch reload fraction, (3) core heterogeneity, or (4) thermal power. The NRC staff finds that the qualification based upon these plants over the transition cycles presented provides adequate assurance that the POLCA7 neutronic method is sufficiently robust to accurately model core critical configuration, and subsequently cold shutdown margin within previously established uncertainties.

References

1-1.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, “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) (Proprietary/Non-proprietary),” LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

1-1.2 CENPD-390-P-A, “The Advanced PHOENIX and POLCA codes for Nuclear Design of Boiling Water Reactors” ABB CE, December 2000. (ADAMS Accession No. ML010100314)

<u>RAI 1-3</u>

The NRC staff requested additional information regarding the qualification of the fuel rod models for high gadolinia loadings characteristic of modern, aggressive, BWR core designs. Westinghouse provided additional information regarding the qualification of the fuel rod thermal conductivity model with higher gadolinia loadings. The POLCA-T fuel rod model is based on STAV7. The fuel thermal conductivity for gadolinia-bearing fuel is modeled

according to an **[**
] in the bulk fuel. The model was compared to
thermal conductivity measurements up to **[** **]** gadolinia. The results of this comparison were provided to the NRC staff in response to RAI 1-3. The results indicate excellent agreement for the conductivity with no apparent degradation in the model capability for fuel conductivity up to **[** **]**

The response further states that the uncertainties **[**
] As the mechanisms of
fuel thermal perform ance models are seldom individually qualified, the NRC staff finds this approach acceptable to capture the net impact of model uncertainties on calculational results.

Therefore, the NRC staff finds that the modeling approach is reasonable based on comparisons to data for **[** **]** Furthermore, the NRC staff's basis for this conclusion is that model **[**

]

Reference

1-3.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

RAI 2: Mixed Cores

RAI 2-1

The NRC staff requested additional information regarding the applicability of the void quality correlation for modern fuel designs (e.g., 10X10 lattices with internal water channels and part-length rods). Westinghouse conducted full scale testing of the **[**
] bundles. These bundles are designed with an **[**
] Therefore, the bundle itself is divided into **[**
]

Testing was performed at the FRIGG thermal-hydraulic test facility. Full-scale **[**
] The measurements were performed using **[**
] The data were compared to measurements for the modern fuel designs as well as older fuel designs **[** **]** The number in the test series refers to
the number of rods in the test bundle. In the case of the **[**
] fuel bundles were tested. The **[** **]** fuel
bundle geometries; therefore, these include modern fuel design features such as part length rods.

The figure provided in response to RAI 2-1 indicates consistent agreement between measurements and predictions over the **[**
] The

test data indicate that the performance of the Dix-Findlay correlation is improved for modern fuel bundle geometries. A statistical summary table provided in the response compares the results of the comparisons for the **[** **]** test separately. The results indicate that that **[** **]** tests, which include **[** **]** measurements, result in a **[**

]

Westinghouse, further, compared many data points to measurements to ascertain if trends were exhibited. The **[**

]

[

]

The **[**

] Experience at Westinghouse-vended EPU and EPU/MELLLA+ cycles **[**

]

The statistical comparison of the void quality correlation to data is based on **[** **]** data points including many data points with modern fuel bundle geometries. As the tests carried out at FRIGG encompass conditions extending to the point of dryout, the NRC staff finds that these tests are reasonably expected to encompass the anticipated operational conditions during normal operation and during anticipated operational occurrences where there is margin to the onset of boiling transition.

Similarly, as no trends are observed in void fraction prediction accuracy for any range of the void measurements, for modern fuel geometries, the NRC staff is reasonably assured that minor extrapolation of the dataset to slightly higher void fractions (**[** **]**) would not result in an appreciable increase in the void fraction uncertainty, given the robustness of the correlation.

The NRC staff finds that the void fraction uncertainty based on the qualification against these modern data is acceptable for quantifying a void fraction uncertainty for downstream analysis of the statistical convolution of uncertainties for steady state and transient calculations for plants operating at the EPU/MELLLA+ condition or lower high-flow control lines.

[

]

References

2-1.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

2-1.2 Westinghouse Presentation on Westinghouse Fuel Performance Update Meeting BWR Fuel Update (Slide Presentation of December 8, 2004), (ADAMS Accession No. ML043580104)

RAI 2-2

The NRC staff requested additional information regarding reload licensing for transition or mixed core analysis. The NRC staff found that review of the reload licensing methodology was outside the scope of the subject TR. Therefore, the NRC staff did not perform a review of the response to RAI 2-2. The NRC staff recommends that the NRC staff review the safety limit determination for specific mixed core applications referencing the subject TR to ensure compliance with the approved methodology.

Reference

2-2.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

RAI 2-3

The NRC staff requested additional information regarding the critical power correlations referenced in the POLCA-T dryout correlation library. As critical heat flux evaluation is outside the scope of the current TR, the NRC staff did not perform a review of the response to RAI 2-3. The NRC staff will defer the review of the response to RAI 2-3 to the review of POLCA-T for either application to transients or to determination of the delta critical power ratio versus oscillation magnitude (DIVOM) slope. Approval of POLCA-T for control rod drop accident (CRDA) analysis or stability analysis will not constitute NRC acceptance of the response to RAI 2-3.

Reference

2-3.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

RAI 2-4

The NRC staff requested additional information regarding the modeling of parallel flow channels within a fuel assembly. Namely, the NRC staff requested that Westinghouse provide additional details of the treatment of sub-bundle flow paths for fuel other than SVEA type fuel assemblies.

The **[**

]

The NRC staff finds that applying this modeling capability in POLCA-T would improve the accuracy of dynamic predictions for modern fuel assemblies. The NRC staff review regarding the dynamic bypass void modeling for stability is addressed in RAI 6-33. Detailed dynamic bypass void formation modeling is not required for CRDA analyses. The NRC staff will review the modeling capability of POLCA-T for dynamic internal bypass channel voiding during its review of POLCA-T for transients.

Reference

2-4.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

<u>RAI 2-5</u>

The NRC staff requested additional information regarding the applicability of the POLCA7/POLCA-T pin power reconstruction methodology for fuel designs other than SVEA types. The response states that the POLCA-T pin power reconstruction method is **[** **]** receiving pin power data from **[** **]** The only inputs into the pin power reconstruction methodology are the **[** **]** The pin power reconstruction method is intended to account for the **[** **]** and therefore, the **[** **]** calculations in POLCA, and the **[** **]** is captured in the PHOENIX **[** **]** Therefore, the process **[** **]** Therefore the NRC staff finds this approach acceptable.

Reference

2-5.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

<u>RAI 2-6</u>

The NRC staff requested additional information regarding the interface between POLCA-T and process computer data for instances where a core is transitioned from another fuel vendor to Westinghouse fuel and fuel licensing analyses. In these cases, the previous cycles were monitored using another core monitoring systems (e.g., 3DMONICORE or POWERPLEX). The NRC staff specifically requested how the process computer fuel data is used to determine legacy

fuel nuclear parameters for licensing analyses. The response states that Westinghouse **[ ]** in these analyses. Instead, Westinghouse performs all calculations based on **[ ]**

Generally, core monitoring systems, including the Westinghouse core monitoring system, use plant data such as **[ ]** to adjust the core simulator results to match plant data and enhance the accuracy of the calculated thermal margins. However, Westinghouse **[ ]**

This practice is common and acceptable **[ ]**

The NRC staff, therefore, finds that the response is acceptable.

Reference

2-6.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

<u>RAIs 2-7 through 2-11 (not used)</u>

<u>RAI 2-12</u>

The NRC staff requested additional information regarding the treatment of subnodal geometric changes. The NRC staff particularly requested this information in the context of mixed cores where several fuel bundles types may be present with rod geometry variations that occur axially throughout the bundle but may occur at axial locations that are inconsistent with the Westinghouse standard nodalization.

The response states that the treatment of bundle thermal-hydraulic data in POLCA-T is consistent with the treatment of bundle thermal-hydraulic data in the approved POLCA7 code. Therefore, the NRC staff notes that the process for collapsing the thermal-hydraulic data has been previously approved by the NRC staff. Additional descriptive details were provided for the precise process of how these data are transferred from POLCA7 to POLCA-T, and the specific details of the treatment of the hydraulic data are also provided in the response (Reference 2-12.1).

The response states that the **[**

]

In terms of modeling the heat transfer, the NRC staff finds that the **[** **]** is sufficiently accurate to explicitly account for characterizing the heat transfer on a nodal level. Since the NRC staff has previously reviewed and approved the POLCA7 nodalization scheme, as well as the treatment of the collapsing of the nuclear parameters for subnodal geometry changes, the NRC staff finds that this approach is acceptable, and consistent with general industry practice in the simulation of reactor system behavior.

In terms of the **[** , **]** the NRC staff agrees with the response that the significant phenomena are treated in terms of accounting for pressure drop. The treatments of the **[** **]** are consistent with the previously approved methodology in POLCA7 and are consistent with general industry practice. Given the relatively **[** **]** the NRC staff has found that the approach is sufficiently accurate in its detail to model transient behavior for mixed core calculations.

Therefore, the NRC staff finds that the response is sufficiently complete in its description and that the modeling approach is acceptable.

Reference

2-12.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Fifth Set of Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-09-25, May 12, 2009. (ADAMS Accession No. ML091380095)

<u>RAI 2-13</u>

The NRC staff requested additional information regarding the implementation of POLCA-T in the reload licensing framework for mixed core applications. The response confirms that CENPD-300-P-A will be revised, and the intention is to incorporate the POLCA-T appendices as optional methods for analysis in the revised TR. The NRC staff finds that this approach is acceptable. The NRC staff, however, will continue to impose the historical POLCA7 mixed fuel condition on POLCA-T. The conditions state that application to non-Westinghouse fuels requires that the NRC be informed of the application and be given the opportunity to review the application.

The NRC staff notes in several RAI responses (see RAI 3) that Westinghouse has provided qualification data regarding the application of their methods to other vendor fuels. These data include the **[** **]** data. Therefore, the NRC staff expects that a letter informing the NRC staff, where applicable, may reference these data as justification for the use of the POLCA-T methods to analyze mixed cores.

Reference

2-13.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Further Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-08-59, December 16, 2008. (ADAMS Accession No. ML083660101)

RAI 3: Expanded Operating Domains

RAIs 3-1, 3-2, 3-3, 3-4, 3-6, 3-7, 3-8, 3-10, and 3-11

In response to RAI 3, Westinghouse provided significant qualification data of the neutronic methods using plant data. Of particular interest to the current application is the core tracking data provided for **[**

] The NRC staff evaluated, specifically, the **[**

]

Figure A.3.1 shows the **[** **]** Figure A.3.2 provides the power/flow operating points during several cycles at the KKL plant where TIP measurements were performed. The TIP measurements were performed for low core flow rates (ranging between 85 percent RCF and 100 percent RCF) and for EPU power levels (as high as 115 percent OLTP). These data provide direct qualification of the Westinghouse neutronic code suite to EPU/MELLLA+ conditions. Figure A.3.3 provides the **[**
] As is evident from the data, there is **[**

]

The PHOENIX4/POLCA7 code system includes robust microscopic nuclide tracking that allows for accurate modeling of nodal neutronic behavior under hard spectrum exposure conditions. Westinghouse provided a plot of the radial and nodal errors as a function of exposure for **[**
] The plot provided in response to RAI 3-7 indicates that there **[**
] in the codes predictive capabilities as a function of the cycle exposure.

Furthermore, Westinghouse compared the results of TIP measurements and gamma scans performed on **[** **]** SVEA fuel bundles in 2002 to the uncertainties reported in CENPD-390-P-A. The comparisons provided in response to RAI 3-4 indicate that the **[**

] Furthermore, Westinghouse provided **[**

]

To address the NRC staff's concerns regarding operation at off-rated conditions, Westinghouse provided detailed comparisons between pre-EPU and EPU core bundle powers, flow rates, void fractions, and spectral index. The NRC staff notes that high energy core designs indicate a hardened neutron spectrum, as evident from the figures shown in the response to RAI 3-3. The NRC staff however, found that the degree of **[** **]** The ranges of core conditions and spectral conditions are within the range of PHOENIX4/POLCA7 calculational capabilities.

To address the NRC staff's concerns regarding the **[**
] The
data is provided by **[**
] The entire cycle was considered at the **[** **]**
The response to RAI 3-11 provides the **[** **]** Control blades

were inserted and withdrawn during the cycle. The **[**

]

Table A.3.1 provides a comparison of the PHOENIX4/POLCA7 nodal and bundle power uncertainties reported in the original licensing topical report to the uncertainties determined from recent measurements at challenging plants. The significant qualification measurements and analyses provide a high degree of reasonable assurance that application of the PHOENIX/POLCA neutronic method to **[**

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References

3-1.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

3-1.2 LTR-NRC-06-57-P, "Draft Slide Presentation for the POLCA-T Topical Report Pre-Submittal Meeting," November 2006. (ADAMS Accession No. ML063000165)

3-1.3 CENPD-390-P-A, "The Advanced PHOENIX and POLCA codes for Nuclear Design of Boiling Water Reactors" ABB CE, December 2000. (ADAMS Accession No. ML010100314)

RAI 3-5

RAI 3-5 requested that Westinghouse 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.

The response explains that the DF01 correlation is the Holmes correlation originally used in the GOBLIN ECCS performance evaluation model. The DF01 application range is the same as the application range quoted for GOBLIN. The DF02 correlation is based **[** **]** and qualified for modern fuel geometries. The response provides data regarding the qualification of the DF02 correlation for modern fuel geometries (SVEA). The tests are described in the response in regards to the pressure, mass flux, heat flux, and power shapes used in the tests.

In particular, the NRC staff reviewed the **[** **]** tests. These tests were performed for the **[** **]** geometries. These tests were performed over a range of normal reactor pressures (70-80 bar) and for a range of mass fluxes and heat fluxes expected during normal operation up to the conditions of critical power. The SF24 test series also included cosine power shapes, which are similar to expected reactor power shapes (as opposed to the uniform power shape for the OF36 test).

The **[**

]

The treatment of uncertainties is documented for the NRC staff's review for each application of POLCA-T. The stability and CRDA analysis methods are qualified based on full scale integral tests. For the case of stability, the uncertainty is established **[** **]** The CRDA analysis uncertainty is based on a **[**

] however,

provides the NRC staff reasonable assurance that the full scale data comparisons do not mask competing effects in the individual models.

When considered with the gamma scan campaigns and TIP measurements referenced in RAI 1 and RAI 3, the NRC staff is reasonably assured that the void quality correlations used in POLCA-T provide adequately accurate predictions when coupled with the neutronic solver.

The response, however, does not fully answer the NRC staff's question. The response contains **[ ]** The NRC staff requested qualification of the heat transfer correlations as described in Section 11 of the subject TR. Westinghouse provided supplemental information in response to this RAI.

Supplemental Information Provided in Response to RAI 3-5 (Audit Open Item 1)

Westinghouse provided a supplement to RAI 3-5 to incorporate qualification data collected in the **[ ]** series of tests. These tests are used to qualify the closure relationships and heat transfer models over a wide range of conditions. The response provides the results of comparisons to both steady state conditions as well as transient conditions to simulate a pressurization transient.

The steady state qualification indicates that the POLCA-T calculations predict **[ ]** The NRC staff finds that this indicates good agreement, particularly noting challenges in the model and measurement uncertainty in the thermocouple data.

The transient pressurization data provided includes fuel bundle conditions reaching dryout and includes prediction of the clad rewet. The temperature predictions were compared to thermocouple data for several nodes in the middle of the test section and towards the top of the test section. The comparisons indicate that the POLCA-T code accurately predicts the **[ ]** and also accurately calculates the temperature rise during dryout. The cladding temperatures in the **[ ]** In each case, the magnitude of the temperature rise is well predicted. The NRC staff provides a figure from the response below for illustrative purposes.

[

]

Figure A.3.4: **[** **]**

The tests are performed post-dryout and include data for the temperature during rewetting. The comparisons between the POLCA-T model and the **[** **]** data indicate that the timing of rewet is **[**

]

However, the NRC staff finds that the measurements and qualification indicate that POLCA-T accurately predicts the heat transfer characteristics of the bundle up to and beyond the point of critical power and is reasonably accurate in the prediction of rewetting. Therefore, the NRC staff finds that the heat transfer correlations are adequately qualified over their full range of anticipated usage.

This information is sufficient to close Audit Open Item 1.

References

3-5.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up 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) (Proprietary/Non-proprietary)," LTR-NRC-07-62, December 12, 2007.

3-5.2 NRC Audit Results Summary Report "WCAP-16747-P: POLCA-T System Analysis Code with Three-Dimensional Core Model", May 2010. (ADAMS Accession No. ML100840432)

3-5.3 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

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). The NRC staff requested, as far as the available data allows, that Westinghouse provide a statistically significant assessment of the POLCA-T code suite to model reactor behavior at low core flow off-rated conditions.

In response to RAI 3-9, Westinghouse provided off-rated startup data. Data provided from startup measurements at **[ ]** indicate that operation at off-normal conditions **[ ]** TIP data was collected over several cycles during the startup with low power and flow conditions. The NRC staff compared the bundle power errors to the uncertainties reported in CENPD-390-P-A. Table A.3.2 summarizes the measurements. The NRC staff compared the bundle errors to the CENPD-390-P-A values and found that the **[ ]** The NRC staff considered the variation with power-to-flow ratio as shown in Figure A.3.5. The NRC staff found that there are **[ ]** Some points are expected to exceed this value as it represents a one-sigma measure of the uncertainty. The NRC staff found that the average of the data sets indicates that the uncertainty **[ ]**

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References

3-9.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

3-9.2 CENPD-390-P-A, "The Advanced PHOENIX and POLCA codes for Nuclear Design of Boiling Water Reactors" ABB CE, December 2000. (ADAMS Accession No. ML010100314)

RAI 4: Code Legacy

RAI 4-1

RAI 4-1 requested that Westinghouse provide a core follow reanalysis of a case contained in the original submittal of POLCA7 to demonstrate that changes made since the original review have not resulted in code drift over time. Code drift is a term referring to code changes within a given code change acceptance criterion over time that result in incremental changes whose net effect is significant relative to the code qualification reviewed and approved by the NRC.

In the original response, the **[**

]

To address the NRC staff's concerns regarding the code uncertainty, Westinghouse provided Reference 4-1.1 for NRC's review during the audit. Reference 5 contains qualification studies of the PHOENIX4/POLCA7 codes against recent data collected in modern cores and critical experiments.

Based on Reference 4-1.1, the PHOENIX4 code was shown to consistently underpredict the fission power in burnable absorber rods. The underprediction is suspected to be the cause for reactivity trends in biases as a function of exposure.

The **[**

]

The NRC staff notes that the POLCA7 accuracy has been quantified using full scale data in a recent production method. The NRC staff, furthermore, notes that the POLCA-T test matrix has been expanded to ensure that its integration into POLCA-T does not adversely affect either stability analyses or complex transients (see Section 3.3.14 of Reference 4-1.2).

The NRC staff also notes that a revision will be made to the final TR to update the numerical results to be consistent with the most recently released version of POLCA7, and that the POLCA-T release consistent with the updated POLCA7 code will be maintained under the

updated POLCA-T Test Matrix. Therefore, the NRC staff finds that the release version described in the final revision to the topical report will be adequately controlled by the quality control process as described in Section 3 of Reference 4-1.2. Therefore, future improvements or changes to the POLCA7 or upstream codes will not adversely affect the reliability of the POLCA-T methodology.

The NRC staff considers the qualification of POLCA7 to adequately provide reasonable assurance that the extension of the neutronic methodology to EPU or EPU/MELLLA+ conditions does not result in an adverse increase in code uncertainty. Therefore, the NRC staff similarly concludes that the POLCA-T kinetics solver's efficacy is likewise unaffected at these conditions.

The NRC staff does not require additional information provided under a separate open item to address the technical concerns associated with RAI 4-1. However, the NRC staff notes that closure of RAI 4-1 relies on information to be provided under separately specified open items, namely the update to the topical report numerical results to reflect the most recent POLCA-T production code (see Sections 3.2.1 and 3.2.2 of Reference 4-1.2).

The NRC staff's evaluation of the Westinghouse response to the audit open items is discussed under RAI 4-11.

References

4-1.1 Casal, J., Krouthen, J., Albendea, M., *Reliable Tools to Model Advanced SVEA Fuel Designs*, ANS 2003, Topical Meeting Advances in Nuclear Fuel Management III. South Carolina, October 5-8 2003.

4-1.2 NRC Audit Results Summary Report "WCAP-16747-P: POLCA-T System Analysis Code with Three-Dimensional Core Model", May 2010. (ADAMS Accession No. ML100840695)

4-1.3 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up 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) (Proprietary/Non-proprietary)," LTR-NRC-07-62, December 12, 2007. (ADAMS Accession No. ML073580493)

<u>RAI 4-2</u>

The NRC staff requested additional information regarding the model descriptions of PHOENIX4/POLCA7. Namely, the NRC staff requested if analyses were used to modify or normalize models in the code system in an empirical or semi-empirical fashion since NRC review and approval of CENPD-390-P-A. The response states that no such modifications were implemented. The NRC staff finds this response acceptable.

Reference

4-2.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

RAI 4-3

The NRC staff requested additional information regarding the branch cases analyzed by PHOENIX4 for downstream use in POLCA7 calculations. Generally, lattice physics calculations are performed for particular void history conditions with branch cases to account for void changes, control rod presence, spacer grid presence, boron concentration, and liquid water temperature. Response to RAI 4-3 provides additional descriptive details of the branch cases and exposure step points. Analyses were provided comparing PHOENX4 to **[**

]

The response is consistent with the reload design procedure used at Westinghouse. During an audit, the NRC staff evaluated the reload design procedure as documented in its audit results summary report (Reference 4-3.2). The NRC staff finds that the branch case selections are reasonable in terms of providing sufficient coverage of reactor operating conditions to establish stable response surface functions for cross sections in the core simulator. The adequacy of the branch case selection procedure is demonstrated by validation of the code suite against relevant gamma scan and TIP measurements described in greater detail in response to RAI 3.

On the basis of demonstrated accuracy of the response surfaces, and its review of the reload design procedure, the NRC staff finds that the approach is acceptable.

References

4-3.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

4-3.2 NRC Audit Results Summary Report "WCAP-16747-P: POLCA-T System Analysis Code with Three-Dimensional Core Model", May 2010. (ADAMS Accession No. ML100840695)

RAI 4-4

The NRC staff requested additional information in regards to the use of POLCA-T uncertainties in establishing safety limits. The response states that the methodology for determining safety limits is based on the CENPD-300-P-A reload licensing evaluation methodology. CENPD-300-P-A describes the analyses and general treatment of uncertainties and conservative assumptions in reload licensing evaluations. POLCA-T specific uncertainties are determined according to the intended application and quantified according to the application specific submittal. In the case of the subject TR these applications are control rod drop accident analysis and the time domain stability application.

The **[**

] Therefore, the response is sufficient in that it states that analysis specific uncertainties and conservatisms are documented in the relevant appendices to the subject TR. As safety limit determination is guided by the generic reload licensing methodology described in CENPD-300-P-A, the NRC staff reiterates that it recommends that the NRC staff review the safety limit determination for specific applications referencing the subject TR to ensure compliance with the approved

methodology, particularly to ensure that the POLCA-T uncertainties developed for each specific application are appropriately accounted for in the reload licensing calculations, appropriate limits, or acceptance criteria.

Reference

4-4.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747P, "POLCA-T: System Analysis Code with Three-Dimensional Core Model" (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

4-4.2 CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel," ABB CE, July 1996 (ADAMS Accession No. ML072250429).

RAI 4-5

The NRC staff requested additional information on how a bundle-specific R-factor is determined. The response states that a specific R-factor is developed for each fuel product line. This response implies that all SVEA-96 Optima2 bundles would have the same R-factor. The NRC staff does not find this to be consistent with the Assembly R-factor calculation, which is dependent on the local pin power distribution. See CENPD-392-P-A as an example.

Westinghouse provided a sample R-factor method in WCAP-16081-P-A. The R-factor is calculated by **[**
] During the audit conducted between March 17, 2008 and March 20, 2008, Westinghouse confirmed that the bundle R-factor is **[**
] is consistent, generally, with the technique described in WCAP-16081-P-A for SVEA-96 Optima2 fuel bundles.

Therefore, the NRC staff finds that the response is adequate and consistent with previously approved R-factor methods.

References

4-5.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

4-5.2 CENPD-392-P-A, "10X10 SVEA Fuel Critical Power Experiments and CPR Correlations: SVEA-96," Combustion Engineering, September 2000. (ADAMS Accession No. ML003767392)

4-5.3 WCAP-16081-P-A, "10x10 SVEA Fuel Critical Power Experiments and CPR Correlation: SVEA-96 Optima2," Westinghouse Electric Company, March 2005. (ADAMS Accession No. ML051260171)

4-5.4 NRC Audit Results Summary Report "WCAP-16747-P: POLCA-T System Analysis Code with Three-Dimensional Core Model", April 2010. (ADAMS Accession No. ML100840695)

RAI 4-6

The NRC staff requested additional information regarding the assignment of direct moderator heat to the bypass and internal water flows (such as in the water cross internal bypass channel). The response states that POLCA-T **[**
] This calculation is performed based on a
[**]** in the POLCA-T model. The direct moderator heating is determined from upstream lattice physics analysis, and POLCA-T treats the internal and external water channels explicitly. Therefore, the NRC staff finds that the methodology is acceptable on the basis that **[**
] and that POLCA-T **[**
]

Reference

4-6.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

RAI 4-7

The NRC staff requested that Westinghouse provide additional information regarding the means for selecting pump homologous curve input for use in POLCA-T. The response states that the data comes from either the pump manufacturer or, in the case of Westinghouse BWR plants, from pump performance data. The NRC staff finds this approach acceptable.

Reference

4-7.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

RAIs 4-8 and 4-9

The NRC staff requested additional information regarding the time step control algorithm. The response states those algorithms that control the minimum time step and specifies that all algorithms are **[** **]** is used in the calculation. The NRC staff generally finds this approach acceptable to ensure adequate time stepping in transient calculations. However, the NRC staff notes that time domain stability analyses are sensitive to the material Courant limit in semi-implicit calculations. Therefore, the NRC staff's review for stability considers additional details regarding the determination of the acceptable time step. The response states that the Courant limit time step control algorithm is optional. Westinghouse provides additional detailed information regarding time step control during time domain stability calculations in response to RAI 6-26.

The NRC staff requested additional information regarding the time step control during calculations for reactivity insertion accidents. The response states that for CRDA, **[**
]

[**]** for the analysis. The response further states that the **[** **]** for time step during such calculations.

The response states that **[**

] The NRC staff finds that such sensitivity studies are acceptable for ensuring adequate time stepping for CRDA analyses. The NRC staff will require that on a plant-specific basis, user-specified time steps be verified using the aforementioned sensitivity analysis technique.

Supplemental Information Provided in Response to RAI 4-8 (Audit Open Item 2)

Westinghouse provided supplemental information regarding the time step selection for stability calculations in Reference 4-8.2. The response states that the stability analyses are performed using a **[** **]** The response further states that this **[** **]** The qualification analyses provided in Appendix B of the TR are consistent with the standard production time step. The NRC staff finds, therefore, that the uncertainty analysis is consistent with the standard production **[** **]** Further, the NRC staff notes that the **[** **]** (see RAIs 6-3 and 6-24). A **[** **]** This approach ensures that the **[** **]** during an analysis. The appropriateness of the node size is provided by the qualification analyses based upon several BWR reactor types. Therefore, the NRC staff finds that the approach is acceptable and adequately justified.

This information is sufficient to close Audit Open Item 2.

References

4-8.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

4-8.2 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Further Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-08-59, December 16, 2008. (ADAMS Accession No. ML083660101)

<u>RAIs 4-10 and 4-11</u>

The NRC staff requested access to the code documentation and to the POLCA-T code itself. The NRC staff evaluated the complete code documentation and performed calculations using POLCA-T during its audit conducted between March 17, 2008 and March 20, 2008.

The relevant findings are documented in the audit results summary report (Reference 4-10.1). The audit is sufficient to close RAI 4-10.

In response to the audit, the NRC staff issued several open items regarding the POLCA-T application to CRDA and stability analysis. The NRC staff requested that Westinghouse address all open items identified during the March 2008, POLCA-T audit in RAI 4-11.

Audit Open Item 1 is addressed in RAI 3-5.

Audit Open Item 2 is addressed in RAI 4-8.

Audit Open Item 3 is addressed in RAI 5-1.

Audit Open Item 4 is addressed in RAI 6-5.

Audit Open Item 5 is addressed in RAI 6-16.

Audit Open Item 6 and Audit Open Item 7 require correction of numerical results in the final revision of the TR to account for code corrections. The NRC staff has reviewed these code corrections as part of its audit (Reference 4-10.1). In Reference 4-10.2, Westinghouse states that the numerical results in the final revision of the TR will be corrected consistent with the code corrections reviewed by the NRC staff as part of the audit.

Audit Open Item 8 is addressed in RAI 6-33.

Audit Open Item 9 is related to the BISON and RAMONA methods and does not apply to POLCA-T. Therefore, the Audit Open Item 9 is addressed outside of the review of POLCA-T. In reference 4-10.2, Westinghouse provided a commitment to submit a follow-up letter to the NRC to resolve Open Item 9. The NRC staff finds that this commitment is acceptable to address the Open Item within the context of the POLCA-T review and defers conclusions regarding the final disposition of Open Item 9 to its review of the follow-up letter.

References

4-10.1 NRC Audit Results Summary Report “WCAP-16747-P: POLCA-T System Analysis Code with Three-Dimensional Core Model”, May 2010. (ADAMS Accession No. ML100840695)

4-10.2 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, “Fifth Set of Responses to the Second Round of NRC’s Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, ‘POLCA-T: System Analysis Code with Three-Dimensional Core Model’ (TAC No. MD5258) (Proprietary/Non-Proprietary),” LTR-NRC-09-25, May 12, 2009. (ADAMS Accession No. ML091380095)

RAI 5: Individual Models and Separate Effects Qualification

RAI 5-1

The response reiterates the testing conditions of the data collected for void-quality correlation qualification. The NRC staff reviewed the void quality correlation testing database and draws conclusions regarding these tests as described in its review of RAI 3-5. The NRC staff specifically requested that Westinghouse provide the results of any transient tests that were performed. The original response refers to the Peach Bottom EOC2 turbine trip tests as an

integral test that provides qualification of the transient application of the void quality correlation. However, as the test essentially compares predicted and measured nuclear instrumentation responses, the NRC staff finds that there are too many interacting nuclear phenomena for the test to provide an adequate demonstration of the transient performance of the correlation.

Supplemental Information Provided for Response to RAI 5-1 (Audit Open Item 3)

Westinghouse provided supplemental information in response to RAI 5-1 to qualify the void quality correlation usage in transient analyses. The response first references the **[ ]** While the **[ ]** does not include specific measurement of the **[ ]** The qualification of the POLCA-T code against the simulated pressurization transient indicates accurate prediction of the **[ ]** During the transient response the code must accurately predict the changes in nodal thermal hydraulic parameters to converge the transient solution. The **[ ]** therefore, provides additional assurance that the void-quality correlation adequately predicts the change in void fraction for transient applications.

The response further provides qualification of the POLCA-T method against the **[ ]** The **[ ]** The test is a full scale integral pump trip test. The test includes individual bundle flow measurements. The test, therefore, allows for the qualification of the POLCA-T code to predict the flow distribution to the bundles ranging from forced circulation to the lowest flow conditions at natural circulation.

The flow is measured in **[ ]** that are located in various positions within the core, which allows for the characterization of the radial flow distribution for bundles near the periphery and bundles near the core center.

The POLCA-T predicted bundle flows were compared against the collected data. The comparison shows a high degree of accuracy of the POLCA-T simulation. The transient reduction in power as well as the transient reduction in flow to natural circulation provides a high degree of confidence in the overall code's predictive capabilities. As the natural circulation flow for each individual bundle is tightly coupled to the nuclear effect of the reduced flow (and hence reduced power) and the thermal-hydraulic effect of the two-phase friction losses within the bundles, the NRC staff finds that it provides reasonable assurance that the void quality correlation can accurately predict the transient change in nodal void conditions for full scale plant applications.

The NRC staff further concludes that the data indicate that the bundle flows are well predicted for each bundle, therefore, indicating that the accuracy of the POLCA-T predicted bundle flows for full scale applications is not dependent on the bundle location.

This information is sufficient to close Audit Open Item 3.

References

5-1.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up 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) (Proprietary/Non-proprietary)," LTR-NRC-07-62, December 12, 2007. (ADAMS Accession No. ML073580493)

5-1.2 NRC Audit Results Summary Report "WCAP-16747-P: POLCA-T System Analysis Code with Three-Dimensional Core Model", May 2010. (ADAMS Accession No. ML100840695)

5-1.3 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No.: ML081890191)

RAI 5-2

The NRC staff requested additional information in regards to the PHOENIX cross section libraries and the **[ ]** The response states that for standard production techniques, **[ ]** is used. The response further states that the **[ ]** reported in CENPD-390-P-A is based originally on an observed bias in the TRX critical experiments. The bias is also seen in the more relevant **[ ]** experiments. The critical experiments referenced in the response are representative of LWR conditions. The NRC staff requested that Westinghouse justify the applicability of the uranium resonance correction factor to other fuel designs. Since the bias is observed for a wide range of conditions in these critical experiments, then the NRC staff is reasonably assured that it is generically applicable. The NRC staff is reasonably assured that the uranium resonance correction factor is appropriate for legacy fuel based also on the response to RAI 1-2.

In the response to RAI 1-2, Westinghouse provided the results of bundle gamma scans performed at the **[ ]** The bundles scanned include several **[ ]** The GE11 fuel assemblies are 9x9 with part-length rods (Fuel-A in the RAI response). The consistency in the accuracy of the gamma scan campaign results for these fuel assemblies with the SVEA-96 assemblies, as well as the consistency between these gamma scan comparisons with the CENDP-390-P-A TR results indicates that the **[ ]** factor is acceptable for modeling a large variety of fuel assemblies including non-Westinghouse fuel assemblies. Therefore, the NRC staff is reasonably assured that this correction factor is adequate to support the uncertainty basis in CENPD-390-P-A for modern fuel designs.

References

5-2.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

5-2.2 CENPD-390-P-A, "The Advanced PHOENIX and POLCA codes for Nuclear Design of Boiling Water Reactors" ABB CE, December 2000. (ADAMS Accession No. ML010100314)

RAI 5-3

The NRC staff requested that Westinghouse provide additional qualification data if approval is sought for POLCA-T application to cores with mixed oxide (MOX) fuel. The response states that Westinghouse is not seeking approval of POLCA-T for application to MOX-fueled cores. Therefore, the NRC staff does not require the requested information to complete its review of POLCA-T and will impose the restriction that POLCA-T is not approved for application to MOX-fueled cores.

Reference

5-3.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

RAIs 5-4 and 5-5

In RAI 5-4 and RAI 5-5, the NRC staff requested additional information regarding the ability of POLCA-T to calculate the minimum critical power ratio (MCPR). Westinghouse responded stating that the scope of the current application of POLCA-T does not include determination of thermal margin, and this information is not required. The NRC staff agrees, as the application of POLCA-T for stability does not include an application to determine the slope of the DIVOM curve for Option III analyses. Therefore, the NRC staff does not require the requested information to complete its review based on the scope of the stability application. The NRC staff, however, does not approve use of POLCA-T to calculate DIVOM parameters and interprets the response as a commitment by Westinghouse to provide the requested information for any application of POLCA-T to analyze anticipated operational occurrences or to calculate DIVOM slope.

Reference

5-4.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

RAI 5-6

The NRC staff requested that Westinghouse provide the details of the calculational method for determining the gamma-smeared pin power distribution. The gamma-smeared pin power distribution accounts for the heat deposited in the fuel and cladding as a result of gamma radiation. Generally, fission energy released in the form of gamma radiation in one pin may be deposited in other pins, thereby resulting in a distribution of pin heat flux that is different from the distribution of pin fission density. The gamma-smearing of the pin powers is done in the **[** **]** according to the response. The calculation is performed in several steps.

[

]

The NRC staff finds that this approach is acceptable for calculating the gamma-smeared pin power distribution.

Reference

5-6.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

RAI 5-7

The NRC staff requested additional information regarding the determination of the uncertainty and biases in the void reactivity coefficient. The response states that information regarding the methods' accuracy for void conditions has been provided in the response to RAIs 1 and 2. The NRC staff further requested that Westinghouse describe how uncertainties and biases in the void reactivity coefficient are captured in safety or operational limits. The response appears to the reference the **[**

]

That NRC staff finds the response acceptable insofar as it specifies that **[**

] The NRC staff notes that for CRDA analyses, the core conditions are **[**

] during the analysis. Therefore, the **[**

] the CRDA results.

Void reactivity feedback, however, plays a role in BWR power/flow oscillations. The NRC staff notes however, that **[** **]** based on the qualification of the POLCA-T method against **[**

] Therefore, the qualification **[**

]

The NRC staff's acceptance of this response to RAI 5-7 should not be construed as the NRC staff's acceptance of the response for all POLCA-T applications. In particular the NRC staff will request similar information for subsequent POLCA-T applications if Appendices C and D are submitted for review and approval.

Reference

5-7.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

RAI 5-8

The NRC staff requested that Westinghouse provide the method for calculation of the non-condensable gas mass, volume, and partial pressure. The NRC staff notes that details of this particular model are not required for the NRC staff to complete its review of the CRDA and stability applications of POLCA-T. The NRC staff, however, is performing, to the extent possible, a generic review of the models in POLCA-T for the host of applications Westinghouse has described in Section 1 of the subject TR.

According to the RAI response, the partial pressure of the non-condensable gases is one of the eight state variables describing a volume cell in POLCA-T. The key features of the model include the model for predicting the distribution of the non-condensable gases in either the liquid or vapor phase. The **[**

]

The response provides sufficient detail to explain how the mass of non-condensables is calculated from the primary cell variables. Therefore, the NRC staff finds the response to provide an acceptably complete description of the model. The NRC staff finds that the use of Henry's law is acceptable for this purpose, and therefore, finds that the model is acceptable.

Reference

5-8.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

RAI 5-9

The NRC staff requested that Westinghouse provide experimental qualification of the ability of the POLCA-T code to predict pressure drop in the core. The response provides descriptive details of three tests performed at FRIGG. The tests were performed for **[**

] The test conditions range in pressure from **[**

] The test bundles were

heated, and the **[**
]
The **[**
] Therefore, the NRC staff finds that the pressure drop data indicate good agreement over the application range.

For all three tests, the qualification indicates excellent agreement between the predicted and measured bundle pressure drops for **[**
] The low mass flux tests typically result in high void conditions in the heated test section and serve to provide additional reasonable assurance that the void quality correlation accurately predicts the bundle conditions at high void fraction.

Reference

5-9.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up 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) (Proprietary/Non-proprietary)," LTR-NRC-07-62, December 12, 2007. (ADAMS Accession No. ML072900261)

RAI 5-10

The NRC staff requested that Westinghouse provide descriptive details of the nuclear instrumentation models in POLCA-T. In particular, certain applications may require modeling the instrument response in order to determine the actuation of systems based on these responses (such as a SCRAM initiated by the reactor protection system (RPS) in response to high flux indications of the APRM). The response states that the POLCA7 nuclear instrument response models previously reviewed by the NRC staff are fully incorporated in POLCA-T. Therefore, the NRC staff finds that the response and the approach are acceptable.

Reference

5-10.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

RAI 6: Stability Evaluation

RAI 6-1

The NRC staff requested that Westinghouse provide additional qualification of the steady state solver for cases applicable to stability analysis. The NRC staff specifically requested that the efficacy of the steady state codes be evaluated for potentially limiting reactor conditions of high power density, low flow, and off-rated conditions. The response states that the response to

RAI 1-2 provides details of the extensive ongoing qualification program of the Westinghouse neutronic methods. The NRC staff specifically considered the qualification data provided in the responses to RAIs 1 and 3. In particular the **[ ]** of the PHOENIX4/POLCA7 steady state methodology for modern operating strategies including operation within the MELLLA+ operating domain.

The response to RAI 3-9 in particular addresses off-rated conditions at high power to flow ratios and is relevant to the qualification of the steady state methodology to determine appropriate initial conditions. The **[ ]** data indicate that the PHOENIX4/POLCA7 methodology is acceptable for predicting steady state power and flow at high power-to-flow ratio conditions.

Lastly, the response refers to the qualification method, which is based on a comparison of calculational results to full scale integral tests. Therefore, the uncertainty derived from the figure of merit inherently includes the uncertainties associated with the initialization of the transient response calculation. The NRC staff agrees with this statement and therefore finds the response acceptable.

Reference

6-1.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

<u>RAI 6-2</u>

The NRC staff requested that Westinghouse provide additional information regarding the biases and uncertainties in the void reactivity coefficient as a function of spectrum hardness. The NRC staff in particular is concerned with the application of the transient stability methodology for modern operating strategies and fuel designs. The response states that the responses to RAIs 2-1 and 3-1 provide separate effects qualification of the relevant physical models. The NRC staff has reviewed these qualification data and found that they indicate acceptable performance of the individual models for conditions typical of modern, aggressive BWR core designs.

The response does not address uncertainties and biases in the void coefficient per se for the time domain stability methodology. The response, similar to RAI 6-1, states that the uncertainty in the figure of merit (decay ratio) is determined from a wide range of plant data collected in European BWR plants. The scope of the qualification covers a wide range of plant and fuel designs. The plants also include **[ ]** and the tests were conducted over a wide range of conditions include marginally stable conditions (small margin to the onset of instability) and in some cases, actual instabilities. The NRC staff, therefore, agrees with the response in that the qualification data and bases are adequate to provide an **[ ]** in the figure of merit. However, in its review of the application of POLCA-T to transients, the NRC staff will ask for similar information as in RAI 6-2.

Reference

6-2.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

RAI 6-3 and RAI 6-24

The NRC staff requested additional information regarding the nodalization and time step control for time domain stability analyses. The response states that **[ ]** is used. The NRC staff notes that **[ ]** The response provides a demonstration of the **[ ]**

The response further states that the core model is based on the POLCA7 core model. In these models, the core is divided into **[ ]** The response states that the thermal-hydraulic cells outside the core are **[ ]** This approach ensures consistency in the **[ ]** in the plant model, and therefore, the NRC staff finds that this nodalization approach is acceptable to preclude numerical damping.

The response to RAI 6-24 indicates the degree of numerical damping associated with a **[ ]** time integration technique. The results compare the same analytical solution to the POLCA-T calculations. The results indicate that a **[ ]** is acceptable to preclude unacceptable numerical damping. The response further states that **[ ]** The NRC staff finds that the question of using separate time integration methods for different parts of the core model has been acceptably resolved as this capability is not proposed for the POLCA-T methodology. However, the responses do not provide sufficient details of the standard production analysis techniques for time domain stability evaluations. The NRC staff requested in RAI 6-26 that Westinghouse provide a detailed description of the nodalization, time step control, and semi-implicitness of the time integration technique used in standard production stability analyses.

References

6-3.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

6-3.2 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up 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) (Proprietary/Non-proprietary)," LTR-NRC-07-62, December 12, 2007. (ADAMS Accession No. ML072900261)

6-3.3 Letter from Gresham, J. A. (Westinghouse) to U. S. Nuclear Regulatory Commission, "Update to the Third Set of Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-09-14, Rev. 1, March 2, 2010. (ADAMS Accession No. ML100830375)

RAI 6-4

The NRC staff requested that Westinghouse describe how the results of stability analyses were used in a licensing framework. The response states that the **[** **]** In the current application, approval is not being sought to approve POLCA-T for DIVOM analyses. The DIVOM curve is generated using the NRC-endorsed RAMONA-3B/BISON/SLAVE methodology.

POLCA-T results are used to **[** **]** The NRC staff finds this acceptable. The shape functions are generically determined, and the NRC staff has found these functions to provide adequate margin to instability.

The **[** **]** the NRC staff finds that this approach is acceptable and consistent with the qualification basis.

Therefore, the NRC staff finds that the response and the **[** **]** are acceptable.

Reference

6-4.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up 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) (Proprietary/Non-proprietary)," LTR-NRC-07-62, December 12, 2007. (ADAMS Accession No. ML072900261)

RAI 6-5

The NRC staff requested additional information regarding Figure B.7-2. The figure includes error bars that **[**
] The response states that **[** the error bars are based on a two-dimensional combination of the uncertainties. **]** Specifically, the response states that the **[**
] uncertainties are included in the figure. The stability measurements
[**]** The NRC staff finds, generally, that this approach is acceptable for presenting the data and qualifying the methodology. However, the NRC staff requested that Westinghouse provide additional details of how the measurement uncertainties are determined.

Supplemental Information Provided for Response to RAI 6-5 (Audit Open Item 4)

The supplemental information provided in response to RAI 6-5 specifies the methodology for determining the measurement uncertainty. The **[**
] The measurement uncertainty is generated **[**

] The uncertainty used in the TR is based on a
[**]** Therefore, the NRC staff finds that the uncertainty magnitude is acceptable.

This information is sufficient to close Audit Open Item 4.

References

6-5.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up 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) (Proprietary/Non-proprietary)," LTR-NRC-07-62, December 12, 2007. (ADAMS Accession No. ML073580493)

6-5.2 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up Response to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-36, August 22, 2008. (ADAMS Accession No.: ML082520770)

RAI 6-6

The NRC staff requested additional information regarding the means for treating bypass void. Under natural circulation or low flow conditions, the NRC staff expects for high power density plants that there is the potential for the formation of significant void in the bypass. The presence of these voids in the bypass may impact the ability of the code to effectively calculate the nodal response to thermal-hydraulic instability. The response provides the details of the bypass void model. In general, the response states that POLCA-T includes

[

]

References

6-6.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

6-6.2 NRC Audit Results Summary Report "WCAP-16747-P: POLCA-T System Analysis Code with Three-Dimensional Core Model", May 2010. (ADAMS Accession No. ML100840695)

RAI 6-7

The NRC staff requested that Westinghouse demonstrate that POLCA-T will predict the onset of an instability event at the appropriate reactor conditions. To demonstrate this capability of the code, Westinghouse provided an analysis using the **[** **]** POLCA-T model. The reactor temperature and core flow were maintained using user input while the reactor power **[** **]** At a power of **[**

]

The response, however, includes a **[**

] However, the response provides the NRC staff with reasonable assurance that the POLCA-T methodology will adequately predict transient oscillatory behavior above the limit-cycle oscillation. The NRC staff further relies on qualification against the SVEA-96 Optima2 stability tests. These tests are provided in Section B.6.1 of the subject TR.

In the SVEA-96 Optima2 stability tests, a POLCA-T model is used to **[**

] The combination of the response to RAI 6-7 and the channel stability tests, provides the NRC staff with reasonable assurance that POLCA-T will predict the onset of the instability event under unstable reactor conditions.

Reference

6-7.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

RAI 6-8

The NRC staff requested that Westinghouse describe the capabilities of the POLCA-T methodology to determine decay ratios for plants that are highly stable. In the response, Westinghouse states that qualification of the POLCA-T methodology at **[**

] The response also states that in the regime where the decay ratio is **[**

] The NRC staff agrees with this determination. Likewise, as the stability methodology is intended to evaluate points in the plant operating domain where the plant may be susceptible to instability, the NRC staff finds that the qualification database is sufficient to establish the capabilities of the methodology and its uncertainties under the conditions relevant to its intended usage. Therefore, the NRC staff finds that the response is acceptable.

Reference

6-8.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

RAI 6-9

The NRC staff requested that Westinghouse provide qualification of the thermal-hydraulic solver in POLCA-T to model density wave instability phenomena. The response includes qualification against the **[** **]** The qualification of the POLCA-T-predicted power for the onset of instability indicates agreement with the **[**

] The calculational results agree with the measurements **[** **]** The data provide the NRC staff with adequate reasonable assurance that POLCA-T includes sufficient detail in **[** **]** to calculate density wave thermal-hydraulic instability phenomena.

References

6-9.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up Response to Question 6-9 of the NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-8, February 7, 2008. (ADAMS Accession No. ML080430051)

RAI 6-10

The NRC staff requested that Westinghouse evaluate the sensitivity of the limit-cycle oscillation magnitude to uncertainties in the interfacial shear. The response states that the phenomenon of **[**

]

The NRC staff agrees. However, the response further states that **[**

] The DIVOM curve used to develop detect and suppress solution (DSS) setpoints is predicated on the RAMONA-3B methodology. The NRC staff finds that the response is acceptable insofar as it has delineated the scope of the current application and dispositions the NRC staff's concern regarding the adequacy of predicting the magnitude of the oscillation.

The current analysis of the decay ratio provides a measure of whether oscillations are expected to diverge or to be damped. To this end, the qualification basis provided in the TR is acceptable to account for uncertainties in an integral sense, and these data are adequate to establish an acceptable calculational limit for the decay ratio to provide reasonable assurance that oscillations are damped. Therefore, the NRC staff does not require additional information based on the scope of the current application.

For application of the POLCA-T methodology to transients or DIVOM curve development, the NRC staff will request additional information regarding the sensitivity to interfacial phenomena.

Reference

6-10.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up 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) (Proprietary/Non-proprietary)," LTR-NRC-07-62, December 12, 2007. (ADAMS Accession No. ML073580493)

RAI 6-11

The NRC staff requested additional information regarding the treatment of friction loss coefficients in the stability analysis methodology. The response states that for Westinghouse fuels, the fuel design is described in a specific TR, and the friction factors for the spacers and orifices are approved with the fuel design. Full scale thermal-hydraulic testing is performed to determine the spacer and orifice loss coefficients used in the analyses.

For non-Westinghouse type fuel, such as legacy fuel, the friction factors are provided by the utility. However, to address the NRC staff's concerns regarding the uncertainties in these loss factors, Westinghouse performed a sensitivity analysis. In this analysis, **[** **]** and the decay ratio and the oscillation frequency were calculated. These studies were performed for the **[** **]** The analytical results indicate **[**

]

Therefore, the NRC staff finds that the response is adequate insofar as it specifies the source of the loss coefficients used in the analysis, quantifies the sensitivity of the results of the stability analysis to these uncertainties in a bounding sense, and provides justification of the qualification data used to establish the uncertainty in the decay ratio.

The response states that component uncertainties are captured in the integral qualification approach. The NRC staff finds that based on the **[** **]** and on the basis of the wide range of reactor conditions and fuel types considered in the qualification database, the uncertainty assessment approach for the POLCA-T time domain stability methodology is acceptable.

Reference

6-11.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

RAI 6-12

The NRC staff requested additional information regarding the conditions of the tests that were included in the stability qualification database. The response to RAI 6-12 includes the reactor conditions in terms of power, flow and power shape. These details provide sufficient information for the NRC staff to complete its review of the qualification data set and the relevant decay ratio measurements and calculational results, and the response is therefore acceptable.

Reference

6-12.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

RAIs 6-13 and 6-20

In RAI 6-13 and RAI 6-20, the NRC staff requested additional information regarding the impact of the balance of plant (BOP) on stability behavior and the analysis methodology. In regards to the modeling of the BOP feedback mechanisms, the response states that the

[

] The NRC staff agrees with the response and finds that this is acceptable. However, changing BOP conditions, **[**

]

The response to RAI 6-20 states that the reactor is protected from developing oscillations during transient conditions by an approved DSS long term solution. Therefore, for the purposes of the current application (determining the exclusion region and controlled entry region), the analysis need only consider the stability margin during steady state conditions. The NRC staff agrees with the response and finds that it is acceptable to address the NRC staff's concerns.

Reference

6-13.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up 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) (Proprietary/Non-proprietary)," LTR-NRC-07-62, December 12, 2007. (ADAMS Accession No. ML073580493)

RAI 6-14

The NRC staff requested additional information regarding the sensitivity of the time domain stability methodology to the disturbance (perturbation) used to initiate the transient calculation. To address the NRC staff's concerns, Westinghouse provided sensitivity study results using the **[**

] These cases specifically evaluated specifically the likelihood of perturbations on non-linearity, and noise impacted the numerical results of the analysis.

The analysis results in the response indicate **[**

] Therefore, the NRC staff finds that the response provides adequate justification of the standard production control rod disturbance technique to evaluate plant decay ratio and is acceptable.

Reference

6-14.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

RAI 6-15

The NRC staff requested that Westinghouse provide justification for the extension of the POLCA-T method to plant designs, operating domains, and fuel designs outside the scope of

the qualification data provided in the TR. The response states that the qualification database is extensive in terms of these considerations. The database includes **[**

] The performance of the POLCA-T methodology has been evaluated for all of these designs **[** **]** and no degradation of the code performance has been observed based on fuel design features.

Similarly, the qualification database **[**
] Therefore, the range of qualification includes a wide variety of plant designs, again without indication of analytical biases based on plant design.

Finally, the qualification includes plants operating at high and low power densities, providing the NRC staff with reasonable assurance that the performance of POLCA-T is consistent over the range of aggressive modern BWR operating strategies and domains.

Therefore, the NRC staff finds that the qualification database itself is sufficiently extensive to cover the range of intended application based on modern BWR fuel designs and core operating strategies and is therefore acceptable.

Reference

6-15.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up 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) (Proprietary/Non-proprietary)," LTR-NRC-07-62, December 12, 2007. (ADAMS Accession No. ML073580493)

<u>RAI 6-16</u>

The NRC staff requested additional information regarding the determination of DR acceptance criterion. The response states that **[**

] The NRC staff agrees with the reasoning in the response, however, is not reasonably assured that specifying DR that is **[**

]

Therefore, the NRC staff will impose a condition on the POLCA-T stability methodology that DR acceptance criterion **[**
]

Supplemental Information Provided for Response to RAI 6-16 (Audit Open Item 5).

The supplemental information provided in Reference 6-16.2 revises the decay ratio acceptance criterion. The criterion has been revised to **[**
]

The response to RAI 6-28 provides additional information regarding the determination of the design margin. The intent of RAI 6-28 was to establish how adequate protection against the onset of instability would be assured by performing POLCA-T calculations against the

acceptance criterion. The response to RAI 6-16 provides **[ ]** Any additional margin afforded in the selection of the decay ratio acceptance criterion is at the discretion of the licensee.

This information is sufficient to close Audit Open Item 5.

References

6-16.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

6-16.2 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Third Set of Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: Systems Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-09-14, Rev. 1, March 2, 2010. (ADAMS Accession No. ML100830375)

<u>RAI 6-17 and RAI 6-22</u>

The NRC staff notes that the TR Appendix B qualification studies for the POLCA-T application to stability include many more core-wide oscillation tests than regional-mode oscillation tests. The primary qualification for regional-mode oscillations is the **[ ]**

RAI 6-17 requests that Westinghouse separately evaluate the uncertainty for each instability mode. The original response states that the driving physical phenomena for regional-model and core-wide oscillations are the same, and therefore, the uncertainty in the calculated decay ratio is not expected to vary between the two modes.

In its review of the response, the NRC staff notes that the regional-mode oscillations require modeling of density wave oscillations, similar to core-wide oscillations, in one-dimensional components, and therefore, agrees that the thermal-hydraulic modeling uncertainties are not expected to vary between the two modes.

However, the NRC staff notes that core-wide power oscillations occur with a radial flux shape that is essentially the same as the fundamental radial flux shape. The regional-mode oscillations result in power tilts along the next highest radial flux harmonic, and therefore, necessarily result in greater local radial flux gradients.

The higher flux gradients may result in errors in the prediction of peak bundle or nodal powers due to the constraints of the diffusion theory solution for large regional power oscillations or radial power shape perturbations.

During its audit between March 17, and March 20, 2008, the NRC staff requested that Westinghouse address the potentially increased uncertainty using the POLCA-T kinetics methods for high radial flux tilts. Westinghouse provided additional qualification information. Part of the qualification studies presented included direct comparisons of the PHOENIX4/POLCA7 methods to the LWR-PROTEUS experiment. A subset of these comparisons was also provided in response to RAI 1-1.

The POLCA-T kinetics solver **[** **]** and therefore, demonstrated efficacy of the **[** **]** provides reasonable assurance that the POLCA-T kinetics solver provides accurate radial flux calculations.

The **[** **]** includes many gamma scan campaigns. Gamma scans provide the most accurate radial power distribution benchmarks by allowing comparison of the calculated bundle and sub-bundle power distributions (as opposed to TIP measurements which do not provide measurement of the bundle powers in an instrumented four-bundle set).

Reference 6-17.1 provides specific qualification of the lattice physics methods and core simulator methods against detailed radial power distribution and axial power distribution measurements. In particular, the NRC staff reviewed the results of LWR-PROTEUS qualification of the lattice pin power calculations as well as qualification of **[** **]**

Several measurements were performed at the LWR-PROTEUS test facility at the Paul Scherrer Institute on a 3 x 3 critical area of full-scale SVEA-96 fuel bundles. The critical measurements included pin power measurements for the central fuel bundle. The central fuel bundle pin powers were compared to the PHOENIX4 infinite lattice calculated power distribution with good agreement. Included in the tests, however, were alternative fuel configurations including the insertion of several control rods to suppress power on two sides of the central fuel bundle. In these tests, the comparison of PHOENIX4 calculations to the measurements indicates only a small increase in the pin power distribution **[** **]** for a very steep radial flux tilt across the bundle.

The LWR-PROTEUS tests confirm that the PHOENIX4 lattice parameters are accurately predicted for even large radial power tilts. The PHOENIX4 lattice parameter input to **[** **]** however, is based on infinite calculations and provides the basis for the pin power reconstruction model. The **[** **]** corrects the radial pin power distribution according to the calculated bundle flux tilt.

Gamma scan measurements, performed at the **[** **]** were performed to qualify the **[** **]** According to Reference 6-17.1, some radial tilts were observed for a fresh fuel bundle in the gamma scan campaign. The measurements indicated a tilt relative to the

calculation across the sub-bundle as **[** **]** for one quadrant of the bundle. While this tilt was observed by the gamma scan measurement, the cause of the tilt has not been fully diagnosed but may be due to **[** **]** that was not explicitly measured or modeled.

The qualification of the **[** **]** provides indirect qualification of the intranodal cross section model. The LWR-PROTEUS experiments demonstrated the accuracy of the PHOENIX4 infinite lattice solution. The **[** **]** provide qualification of the **[**

] to characterize the variation in neutron flux across a node, and, therefore, to model steep radial flux gradients as may be present during regional mode oscillations.

The results for a once-burnt fuel assembly indicate excellent agreement between pin gamma scans and the **[** **]** However, for the fresh fuel assembly,

[

]

During regional mode oscillations, the radial power peaking is a combination of peaking due to both the fundamental and first harmonic flux shapes. The gradient in the first harmonic will be slightly greater than the fundamental mode. Qualification of the pin power reconstruction model with a high degree of accuracy for several gamma scans (excluding the fresh bundle scan) as well as accurate representation of the **[**

]

This is further supported by the comparison of the **[** **]** recorded during the regional mode instability test as well as the calculation of the SPERT III E power as presented in the rod drop qualification, which is highly, radially peaked.

The response to RAI 6-22 provides a **[**

]

Supplemental Information Provided for Response to RAI 6-17.

The response states that the application of POLCA-T in the current TR is restricted to the prediction of decay ratios. DR prediction is required for back-up stability protection (BSP) and is differentiated from the modeling required to determine the DIVOM slope. Based on the limited application, the response states that searches are performed for damped oscillations (DR less than unity) for both modes. As the qualification extends to unstable modes, the behavior of the oscillations is expected to scale linearly. The NRC staff agrees with this assertion.

Therefore, the NRC staff has found that **[**

]

Based on the qualification of the fundamental physics models and the limited application of the methodology to damped oscillations, the NRC staff finds that applying the same acceptance criterion is acceptable for the intended applications.

References

6-17.1 Casal, J., Krouthen, J., Albendea, M., *Reliable Tools to Model Advanced SVEA Fuel Designs*, ANS 2003, Topical Meeting Advances in Nuclear Fuel Management III. South Carolina, October 5-8 2003.

6-17.2 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, “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) (Proprietary/Non-proprietary),” LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

6-17.3 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up Response to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-36, August 22, 2008. (ADAMS Accession No. ML082520770)

6-17.4 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up 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) (Proprietary/Non-proprietary)," LTR-NRC-07-62, December 12, 2007. (ADAMS Accession No. ML073580493)

6-17.5 NRC Audit Results Summary Report "WCAP-16747-P: POLCA-T System Analysis Code with Three-Dimensional Core Model", May 2010. (ADAMS Accession No. ML100840695)

RAI 6-18

The NRC staff requested that Westinghouse provide additional details regarding the methodology used to determine the regional mode oscillation symmetry plane. The response states that **[**

]

In the **[**

]

The NRC staff finds that **[** **]** is acceptable for establishing the regional mode oscillation symmetry plane.

Reference

6-18.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up 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) (Proprietary/Non-proprietary)," LTR-NRC-07-62, December 12, 2007. (ADAMS Accession No. ML073580493)

RAI 6-19

The NRC staff requested that Westinghouse demonstrate the effect of numerical damping using a simple problem. The NRC staff specifically requested that Westinghouse evaluate the ability of POLCA-T using standard production methods to track **[**

] The NRC staff finds that this response is acceptable to demonstrate the efficacy of the numerical time integration technique to preclude numerical damping of oscillations.

Reference

6-19.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up 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) (Proprietary/Non-proprietary)," LTR-NRC-07-62, December 12, 2007. (ADAMS Accession No. ML073580493)

RAI 6-20

The NRC staff requested additional information regarding the use of the DR acceptance criterion to protect against exceeding Specified Acceptable Fuel Design Limit (SAFDLs). The response states that the **[**

]

Initiation of a transient at points outside the exclusion region that would result in the reactor approaching unstable conditions will result in the activation of the Option III suppression function, either through an OPRM SCRAM or through the BSP. **[**

]

The NRC staff finds that the response is sufficient to describe how DR methodology is used to protect SAFDLs.

Reference

6-20.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up 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) (Proprietary/Non-proprietary)," LTR-NRC-07-62, December 12, 2007. (ADAMS Accession No. ML073580493)

RAI 6-21

At the onset of a disturbance to the steady state condition, several oscillatory modes may be excited. In particular, the NRC staff requested that Westinghouse evaluate the ability of the POLCA-T methodology to correctly evaluate DR in instances where several oscillation modes are excited by the disturbance. In response to RAI 6-14, Westinghouse provided the results of a

[
] The response to RAI 6-14 indicates that the DR results **[**
]

The response to RAI 6-21 states that **[**

]
The NRC staff finds that the approach is reasonable to confirm that **[**
] The resilience of the method was explicitly demonstrated in the numerical results of analyses provided in response to RAI 6-14.

References

6-21.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up 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) (Proprietary/Non-proprietary)," LTR-NRC-07-62, December 12, 2007. (ADAMS Accession No. ML073580493)

6-21.2 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

RAI 6-22

The NRC staff requested additional information regarding the predictive capabilities of POLCA-T to determine the azimuthal flux harmonic. The response provides details regarding the qualification of the POLCA7 and POLCA-T codes to predict the radial power distribution during the **[** **]** The results indicate that the POLCA neutronic methods are accurate in the prediction of the regional mode flux shape. Therefore, the NRC staff finds that the models are acceptable for predicting the regional mode flux shapes.

The NRC staff also requested information regarding the specific evaluations performed for Option III plants. The response states that approval is only sought to determine decay ratios. Decay ratio evaluations are performed for several long term stability solution (LTS) options. Generally, the decay ratios are used to generate exclusion regions based on the BWROG LTS methodology. The response states that **[**
] These historical methods have been previously approved by the NRC staff. Therefore, the NRC staff review did not consider the application of POLCA-T to stability analyses other than predicting DR for regional, core-wide, and channel modes.

Reference

6-22.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up 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) (Proprietary/Non-proprietary)," LTR-NRC-07-62, December 12, 2007. (ADAMS Accession No. ML073580493)

RAI 6-23

The NRC staff requested additional information regarding the sensitivity of the transient results to the frequency of performing nuclear calculations during thermal-hydraulic calculations. The response states that **[**
] The NRC staff finds that the **[** **]** and the NRC staff's concern is adequately resolved by the response.

Reference

6-23.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up 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) (Proprietary/Non-proprietary)," LTR-NRC-07-62, December 12, 2007. (ADAMS Accession No. ML073580493)

RAI 6-25

The NRC staff requested additional information regarding the use of POLCA-T analysis results to implement approved stability licensing methods. In particular, the NRC staff requested that Westinghouse describe the processes by which POLCA-T analyses are integrated into the licensing approach for BWROG LTSs. The response provided in Reference 6-25.1 describes the reload licensing analysis procedures for the common long term stability solutions. These solutions are described in References 6-25.2, 6-25.3, 6-25.4, and 6-25.5.

The response describes a general reload analysis procedure that is used to determine (1) limiting operating points within the domain and (2) power-flow stability boundaries. These procedures involve the calculation of decay ratios using the POLCA-T methodology.

Enhanced Option I-A

For plants implementing the Enhanced Option I-A (EIA), the reload licensing analyses are performed in accordance with the NRC-approved licensing topical report NEDO-32339 (Reference 6-25.2). NEDO-32339 defines the process of determining the exclusion, restricted, and monitored regions in the power-flow map. These procedures are approved and are directly adopted for plant licensing. DR acceptance limits are provided in the acceptance criterion for WCAP-16747-P, which is provided in the response to RAI 6-16 (Reference 6-25.1).

On the basis that approved procedures are employed and that the POLCA-T acceptance limits are adopted in the methodology, the NRC staff finds the response in terms of EIA to be acceptable.

Option I-D

The Option I-D LTS is intended for application to small BWR cores with tight inlet orifices. These plant designs are not susceptible to regional mode oscillations. When applying the POLCA-T methods to Option I-D plants, the response provides the analysis procedures. The POLCA-T code is used to calculate decay ratios to support the **[**

]

The response adequately describes how POLCA-T DR results are used within the licensing framework for Option I-D. Therefore, the NRC staff finds that the response is acceptable.

Option II

The Option II LTS is designed for application to BWR/2 plant designs where the quadrant symmetric-based APRM is sufficient to actuate suppression of regional-mode power oscillations directly. In the licensing framework, POLCA-T stability calculations are used to
[ ]

The response adequately describes how POLCA-T DR results are used with the licensing framework for Option II. Therefore, the NRC staff finds that the response is acceptable.

Option III Back-up Stability Protection (BSP)

The Option III LTS is a detect and suppress solution. Option III protects the fuel from exceeding SAFDLs during power oscillations by initiating a reactor SCRAM through the oscillation power range monitor (OPRM). One part of the implementation of the Option III solution is the cycle-specific analysis to determine the OPRM setpoint. POLCA-T methods are not used in the process. **[ ]**

Option III includes Back-up Stability Protection (BSP) for instances when the OPRM is unavailable. **[ ]** Therefore, the NRC staff finds that the response addresses the usage of POLCA-T analysis results within the Option III licensing framework.

The response adequately describes how POLCA-T DR results are used with the licensing framework for Option III. The NRC staff finds that the usage is limited to those applications (DR calculations) approved by the NRC staff herein. Therefore, the NRC staff finds that the response is acceptable.

References

6-25.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, “Update to the Third Set of Responses to the Second Round of NRC’s Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, ‘POLCA-T: Systems Analysis Code with Three-Dimensional Core Model’ (TAC No. MD5258) (Proprietary/Non-Proprietary),” LTR-NRC-09-14, Rev. 1, March 2, 2010. (ADAMS Accession No. ML1008375)

6-25.2 NEDO-32339-A, Revision 1, “Reactor Stability Long Term Solution: Enhanced Option I-A,” April 1998.

6-25.3 NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," August 1996.

6-25.4 NEDO-31960-A, "BWR Owners’ Group. Long Term Stability Solutions Licensing Methodology," November 1995.

6-25.5 NEDO-31960-A, Supplement 1, “BWR Owners’ Group Long Term Stability Solutions Licensing Methodology (Supplement 1),” November 1995.

RAI 6-26

The NRC staff requested additional information regarding the selection of an appropriate axial nodalization. The response provided in Reference 6-26.1 states that the **[**

]

The response references the nodalization and time step information provided to the NRC staff in response to RAI 4-8 and RAI 6-3. The response also states that the time step and axial nodalization for standard production analyses are the same as those employed in the integral qualification analyses to ensure that the uncertainties determined from the qualification remain applicable.

The NRC staff finds that the response is acceptable insofar as it adequately justifies the selection of the node size and time step for the stability calculations performed with POLCA-T. The usage of consistent inputs in this regard is required to justify the applicability of the prediction uncertainty provided in the TR. Therefore, the NRC staff will impose conditions on the application of POLCA-T to perform stability licensing evaluations in terms of the time step and nodalization.

Reference

6-26.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, “Update to the Third Set of Responses to the Second Round of NRC’s Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, ‘POLCA-T: Systems Analysis Code with Three-Dimensional Core Model’ (TAC No. MD5258) (Proprietary/Non-Proprietary),” LTR-NRC-09-14, Rev.1, March 2, 2010. (ADAMS Accession No. ML1008375)

RAI 6-27

The NRC staff requested additional information regarding the applicability of the prediction uncertainty. Specifically, the NRC staff requested that Westinghouse evaluate any additional uncertainty that may be introduced when inputs are specified consistent with the standard production analysis procedure relative to the specification that was used for the qualification analyses presented in the TR.

The response provided in Reference 6-27.1 states that the qualification analysis procedures and the standard production procedures **[** **]** Therefore, the uncertainties in code inputs for the standard production procedure are **[** **]** These include the inputs such as time step size and nodalization. On the basis that the **[**

]

Therefore, the NRC staff finds that the response is acceptable.

Reference

6-27.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Update to the Third Set of Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: Systems Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-09-14, Rev. 1 March 2, 2010. (ADAMS Accession No. ML1008375)

RAI 6-28

The NRC staff requested additional information regarding the decay ratio acceptance criterion. Specifically, the NRC staff notes that the sensitivity to particular plant parameters may require a reduction in the decay ratio acceptance criterion. The response provided in Reference 6-28.1 states that standard uncertainties (generic or plant-specific) for parameters identified in the phenomena identification and ranking table (PIRT) are applied as part of the reload analysis and incorporated in the design margin.

The response states that this is performed on a cycle-specific basis. However, to assist the NRC staff in its review of the reload safety analysis methodology, Westinghouse provided an example analysis to assess the design margin for **[ ]**

The NRC staff reviewed the example analysis and found that appropriate plant uncertainties were considered. The NRC staff has reviewed the stability PIRT that was provided in the response to RAI 6-33 and found that the PIRT is acceptable.

The response further provides a summary of the current reload analysis procedure used for RAMONA-3B stability calculations. This approach is likewise adopted for POLCA-T. In the procedures, the reload safety analysis **[**

]

The NRC staff considered the response to RAI 6-28 in light of the information provided in the response to RAI 6-16. DR acceptance criterion is now defined as **[**
] In response to RAI 6-16,
[
] The response to RAI 6-16 states that in
[
]

The choice to adopt a more conservative decay ratio acceptance criterion remains the discretion of the licensee. The NRC staff requires that the **[**
] Insofar as the revised criterion in RAI 6-16 meets the NRC staff's requirements, the NRC staff finds that the response provides adequate clarification of the use of design margin in Westinghouse reload safety analyses and is therefore acceptable.

Reference

6-28.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Update to the Third Set of Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: Systems Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-09-14, Rev.1, March 2, 2010. (ADAMS Accession No. ML1008375)

RAI 6-29

The NRC staff requested additional information regarding the process by which a regional mode oscillation is established in POLCA-T for cases where the global oscillation mode is dominant.

The response provided in Reference 6-29.1 states that in this case, pressure boundary conditions are applied to the upper and lower plena. These boundary conditions ensure that the **[**

]

The response states that **[**

] The NRC staff finds this approach acceptable. The response further states that **[**

] A sample calculation was provided in the response as an illustration of the **[**

]

This methodology is the **[**

] Therefore, the NRC staff finds that the approach is acceptable. In its review of this approach, the NRC staff states specific requirements for the usage of this method in its safety evaluation attached to CENPD-295-P-A (Reference 6-29.2). The NRC staff finds that these conditions for the instability-threshold power calculations are likewise applicable to POLCA-T.

References

6-29.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Update to the Third Set of Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: Systems Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-09-14, Rev. 1 March 13, 2010. (ADAMS Accession No. ML1008375)

6-29.2 CENPD-295-P-A, "Thermal-Hydraulic Stability Methodology for Boiling Water Reactors," ABB CE, July 1996.

RAI 6-30

The NRC staff requested that Westinghouse perform a sensitivity study to demonstrate that the code captures local effects consistent with other stability codes in the assessment of mixed core designs. In particular, the NRC staff's request for additional information references calculations performed by the NRC staff using the LAPUR frequency domain stability code and calculations that were performed by Westinghouse using the RAMONA-3B code to simulate the

[**]** instability event.

Calculations performed using the LAPUR and RAMONA-3B codes predicted similar mixed core effects that demonstrate the sensitivity of the stability performance to differences in the relative void reactivity and density wave oscillation stability characteristics for different co-resident fuel designs.

The response provided in Reference 6-30.1 states that the **[** **]** core model was not available to Westinghouse; however, sensitivity studies were performed using the **[** **]** This model is a **[** **]** The response provides the results of the sensitivity study to demonstrate that POLCA-T consistently captures the relevant phenomena. The response states that the variation in the decay ratio for **[** **]** This is consistent with the variation observed in the previous sensitivity studies referenced by the NRC staff in the original RAI.

Therefore, the NRC staff finds that the performance of POLCA-T is consistent with the NRC staff's expectations based on similar sensitivity studies. The consistent performance provides the NRC staff with reasonable assurance that the POLCA-T code is adequately modeling the appropriate phenomena affected by the fuel bundle design. On these bases, the NRC staff finds that the response is acceptable.

Reference

6-30.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Update to the Third Set of Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: Systems Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-09-14, Rev. 1, March 2, 2010. (ADAMS Accession No. ML1008375)

<u>RAI 6-31</u>

To assist the NRC staff in its review of the validation information, the NRC staff requested that Westinghouse evaluate the measured and calculated DR data against a parameter selected by the NRC staff as representative of anticipated trends in stability margin. The NRC staff provided a dimensionless parameter that captures dominant reactor conditions affecting stability. This dimensionless parameter is equivalent to a parameter considered in the qualification of the RAMONA-3B code as reported in Figure 7.2-7 of CENPD-294-P-A.

The POLCA-T results are **[** **]** The NRC staff requested that Westinghouse compare the trend lines through the measured and calculated values. The NRC staff understands that this does not provide for the determination of uncertainties, however, the NRC stresses that the comparison is valuable in terms of demonstrating that POLCA-T predicts that the influence of important core parameters on overall stability performance is consistent with observations based on data.

The comparison of the trend lines reveals that they are **[** **]** Therefore, the NRC staff finds that the comparison provides reasonable assurance that POLCA-T predictions of important phenomena are consistent with experimental observations. On this basis, the NRC staff finds that the response is acceptable.

References

6-31.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Update to the Third Set of Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: Systems Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-09-14, March 2, 2010. (ADAMS Accession No. ML1008375)

6-31.2 CENPD-294-P-A, "Thermal-Hydraulic Stability Methods for Boiling Water Reactors," ABB CE, July 1996.

RAI 6-32

Parts A, B, and D

The NRC staff requested additional information regarding comparisons of the POLCA-T and RAMONA-3B codes against the same qualification data collected at **[ ]** Several **[ ]**

The NRC staff requested that Westinghouse provide qualification data against the **[ ]** The NRC staff specifically requested that Westinghouse provide details of the calculation methodology, the calculated mass flow rates, and the calculated LPRM indications at symmetric locations.

The response to RAI 6-32 provided in Reference 6-32.1 states that the **[ ]** The response provides a figure showing the **[ ]** was initiated in the calculation. The NRC staff finds that the **[ ]** are selected to be consistent with the measured harmonic symmetry plane, and are therefore appropriate for exciting the regional mode oscillation.

The NRC staff specifically requested that Westinghouse provide these qualification calculations for the **[ ]**

Plots of the calculated mass flow rates in the bundles surrounding symmetric LPRM locations indicate **[ ]** These calculations confirm that the POLCA-T code is predicting consistent regional oscillatory behavior consistent with the physical behavior observed at **[ ]**

The NRC staff requested that Westinghouse compare the calculated POLCA-T LPRM indications with those calculated by RAMONA-3B. The response states that these methodologies are significantly different and that the comparison adds little value. However, the response provides the information requested by the NRC staff. The NRC staff requested this information to understand how differences in the methodologies affect the accuracy of either code to simulate regional mode oscillations.

The NRC staff reviewed the comparison of the RAMONA-3B and POLCA-T calculations. For the specific LPRM data, the calculations are in good agreement in terms of **[**

]

The NRC staff requested additional information regarding comparison of the void propagation time and the oscillation frequency. The oscillation frequency is driven by the void propagation time through the core. The response includes a comparison of the void propagation time and the frequency. The **[**

]

On the basis of the excellent agreement between transient POLCA-T predicted and measured LPRM signals, and excellent agreement in the regional mode decay ratio between POLCA-T and the **[** **]** measurement, the NRC staff is reasonably assured that POLCA-T adequately models regional mode oscillations.

In Part D of the RAI, the NRC staff requested that Westinghouse compare the void quality correlations. The response clarifies the origin of the RAMONA-3B slip correlation, and states that the requested information was effectively provided in the response to RAI 8-5 (Reference 6-32.2). The NRC staff agrees with the response, and the NRC staff's review of the response to RAI 8-5 is documented separately in this Appendix.

<u>Part C</u>

The NRC staff requested that Westinghouse provide additional information regarding the selection of an appropriate perturbation to excite regional mode oscillations in Reference 6-32.1. The NRC staff requested that Westinghouse address the potential to excite higher harmonic modes with small eigenvalue differences.

The response to Part C of the RAI states that eigenvalue separation is **[**

] The NRC staff notes that imposing a large perturbation may excite several oscillatory modes, and therefore, when observing the transient response the decay ratio may be **[**

]

The Westinghouse methodology described in the response to Part C addresses the NRC staff concern regarding the predicted DR. In the methodology proposed, the **[**

]

The Westinghouse methodology also considers the **[**

] This is addressed in

RAI 6-18. In the response to RAI 6-18 (Reference 6-32.3), Westinghouse **[**

] were provided in the response to RAI 6-29 (Reference 6-32.1).

The NRC staff finds that the responses to RAI 6-32 Part C, RAI 6-18, and RAI 6-29 provide an acceptable methodology for analyzing regional mode oscillations.

References

6-32.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Third Set of Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: Systems Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-09-14, Rev. 1, March 2, 2010. (ADAMS Accession No. ML100830375)

6-32.2 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up Response to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-36, August 22, 2008. (ADAMS Accession No. ML082520770)

6-32.3 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up 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) (Proprietary/Non-proprietary)," LTR-NRC-07-62, December 12, 2007. (ADAMS Accession No. ML073580493)

RAI 6-33

The NRC staff requested that Westinghouse provide a stability PIRT. As not all thermal-hydraulic, mechanical, and neutronic phenomena affect particular analyses in equal ways, the PIRT provides a method for identifying the key physical phenomena that affect particular safety analyses. The NRC staff requested the PIRT to assist the NRC staff in establishing that POLCA-T has adequate capabilities to model those phenomena that are highly important in the evaluation of stability.

The PIRT provided by Westinghouse in the RAI response ranks the phenomena for each of the stability analyses described in Appendix B of the TR. These include the calculation of the global, regional, and channel instability modes. The designators: L (low), M (medium), and H (high) rank the importance of the phenomena listed in the PIRT. The PIRT is subdivided into several subcategories. The NRC staff reviewed the ranking of the phenomena in these subcategories separately.

Rankings

[ ]

The NRC staff has reviewed the ranking provided in the PIRT. The NRC staff concurs with the rankings for most of the phenomena listed in the table. The NRC staff identified certain phenomena where clarification is required for the rationale for the ranking. The **[**

] While the NRC staff considers this phenomenon to be of low importance, the NRC staff notes that when developing a PIRT, it is generally conservative to rank a phenomenon as a higher degree of importance.

Westinghouse has, however, ranked the **[**
] as low in the PIRT. These **[**

] The NRC staff concurs with ranking these phenomena as high because they directly affect the **[**
]

During an on-site audit at Westinghouse's Energy Center, the NRC staff identified an open item regarding the determination of the **[**

] will likely have a greater effect. The NRC staff does not find that the current rationale is sufficient for the NRC staff to be reasonably assured of the ranking of low.

The NRC staff requested supplemental information regarding the PIRT in RAI 6-33S1 pertaining to the **[**

] Therefore, the NRC staff agrees with these rankings.

The **[**

] Therefore, the NRC staff agrees with the low ranking of this phenomenon.

[

] the NRC staff concurs with a rank of high for these phenomena.

[ ]

The NRC staff has reviewed the PIRT rankings under this subcategory and agrees with the Westinghouse rankings. The NRC staff reviewed the medium-ranked PIRT regarding the **[ ]** and the ranking is appropriate. The NRC staff considered the ranking for the **[ ]** and agrees that it is a highly ranked phenomenon insofar as **[ ]**

The Westinghouse PIRT ranks **[ ]** as low ranked phenomena besides the **[ ]** which is ranked as medium. The NRC staff agrees with the assessment. The **[ ]** was ranked medium insofar as the **[ ]** The NRC staff agrees with the rationale and the ranking.

[ ]

The PIRT correctly identifies **[ ]** as a highly ranked phenomenon for **[ ]** therefore, these parameters are ranked as low. The NRC staff agrees with the distinction for these phenomena.

The **[ ]** are ranked as high. The **[ ]** Therefore, the NRC staff concurs with these rankings.

The PIRT likewise ranks **[ ]** as highly ranked phenomena for the global and regional-modes. The NRC staff agrees with these rankings. The **[ ]** will enhance neutronic feedback.

The parameters affecting the **[ ]** were also ranked as highly important phenomena. The **[ ]**

Capabilities Assessment

The initial conditions in POLCA-T are **[** **]** Therefore, the NRC staff reviewed the POLCA-T capabilities in terms of the highly ranked PIRT phenomena identified in the **[** **]** Therefore, the NRC staff concludes that POLCA-T has sufficient capability relative to this highly ranked PIRT.

[**]**

The NRC staff requested additional information regarding the **[** **]** Westinghouse, in these RAI responses, provided the data references **[** **]** In these responses, Westinghouse justified the **[** **]** The NRC staff concludes that POLCA-T has sufficient capability relative to this highly ranked PIRT.

[**]**

The NRC staff has requested additional information in RAI 6-33S1 regarding the **[** **]** The NRC staff has reviewed the **[** **]** The NRC staff notes that it is conservative to bias the heat resistance high for CRDA calculations, but the inverse is true for stability evaluations, as the reactor will be less stable under conditions when the neutronic power response and fluid conditions are more tightly coupled.

The NRC staff reviewed the **[** **]** provided in Appendix B of the TR noting the NRC staff conclusions regarding the **[** **]** The NRC staff found that the **[** **]** Therefore, the NRC staff concludes that the **[** **]** The overall impact is a self-compensating effect.

The **[**

]
Therefore, the NRC staff concludes that the **[** **]** in POLCA-T is sufficiently accurate **[**
] is not compromised.

Therefore, the NRC staff finds that the **[**
] is sufficiently robust for stability evaluations.

[

] in POLCA-T is based on an acceptable **[**
] Therefore, the NRC staff finds that the POLCA-T capabilities are sufficient relative to this highly ranked PIRT.

[

] The responses to these RAIs provide adequate demonstration of the capabilities of POLCA-T relative to this highly ranked PIRT.

[

] the NRC staff requested that Westinghouse justify the POLCA-T capability in this regard. The NRC staff imposes the condition that for **[**

] The responses to these RAIs are sufficient to demonstrate the capabilities of POLCA-T relative to this highly ranked PIRT.

[

] are sufficient to demonstrate the capabilities of POLCA-T relative to these highly ranked PIRT as the **[**

]

Westinghouse provided **[**
] The response to RAI 8-8 is sufficient to demonstrate the capabilities of POLCA-T relative to this highly ranked PIRT.

[

] Therefore, POLCA-T has sufficient capability relative to this highly ranked PIRT.

<u>Supplemental Information Provided for Response to RAI 6-33S1</u>

[**]**

The response to RAI 6-33S1 Part 1 states that the **[**

] The response justifies the rationale by providing the results of sensitivity analyses performed by perturbing the associated parameters (Reference 6-33.2).

As a basis for comparison, the sensitivity to the **[**

]
Therefore, the NRC staff agrees with the rationale for the low ranking of these phenomena in the PIRT.

As a comparison, the sensitivity to the **[**

] This degree of variation is consistent with the NRC staff's expectations in terms of the sensitivity of the stability analysis to a key parameter **[**
]

Since the primary phenomenon affected by these parameters is the **[**

] Therefore, the NRC staff agrees that it is not inconsistent to rank the **[**

]
Therefore, on a similar basis, the NRC staff finds that the PIRT ranking for the **[** **]** is appropriate.

[

] are the subject of Audit Open Item 8, and, by reference, RAI 4-11. The NRC staff requested in RAI 6-33S1 that Westinghouse provide similar details of the **[**
]

The response provides descriptive details of the **[** **]** (Reference 6-33.2). The **[**

] The NRC staff has reviewed these individual models as described in the TR and found them to be acceptable for the purpose of evaluating these phenomena. Therefore, the NRC staff finds that the response is acceptable.

In terms of the **[**

] that has been deferred to future POLCA-T submittals.

In the case of **[** **]** the response to Part 1 of the RAI characterizes the **[**

] Therefore, the NRC staff finds that it is acceptable for the current purposes to **[** **]**

Part 2 of the RAI response provides sufficient information regarding the treatment of the **[** **]** to acceptably resolve Audit Open Item 8.

This information is sufficient to close Audit Open Item 8.

[ ]

The response to RAI 6-33S1 Part 3 provides the methodology employed by Westinghouse to account for **[ ]** (Reference 6-33.2). The methodology relies on the analysis of a **[ ]**

This methodology is acceptable to conservatively determine the **[ ]**

Therefore, the NRC staff finds that the methodology is acceptable. However, the NRC staff will require that the likelihood of significant bypass void formation be assessed for regional mode analyses. Similarly, the NRC staff will require that the analyses documenting the effect of dynamic bypass void formation be included in the Westinghouse Reload Safety Evaluation (WRSE).

References

6-33.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Further Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-08-59, December 16, 2008. (ADAMS Accession No. ML083660101)

6-33.2 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Fourth Set of Responses to the Second Round of NRC's Request of Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-09-21, April 8, 2009. (ADAMS Accession No. ML100281005)

6-33.3 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Third Set of Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: Systems Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-09-14, Rev. 1, March 2, 2010 (ADAMS Accession No. ML100830375)

RAI 6-34

The NRC staff requested that Westinghouse use data from the Peach Bottom 2 turbine trip (PB2 TT) tests to address the sensitivity of pressure wave propagation-to-nodalization as a means for assessing the impact of nodalization on density wave oscillation simulation. The response states that sufficient information is provided in the integral qualification. The NRC staff agrees that the integral qualification provides assurances that the methodology is accurately predicting density wave phenomena.

The NRC staff has requested additional information regarding the effects of nodalization in RAI 4-8, RAI 6-3, RAI 6-19, RAI 6-24, RAI 6-26, and RAI 6-27. The NRC staff reviewed these RAIs, and the detailed findings are provided in this Appendix. The NRC staff found that when these responses are considered that additional information regarding the influence of nodalization on the stability calculations is not required. Therefore, the NRC staff finds that the concern associated with RAI 6-34 has been adequately resolved by information provided in other RAI responses.

Reference

6-34.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Third Set of Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: Systems Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-09-14, Rev.1, March 2, 2010. (ADAMS Accession No. ML100830375)

RAI 6-35

The NRC staff requested that Westinghouse provide the nominal flow rate for the **[ ]** plant in absolute units. The response to RAI 6-35 provided the flow rate. The NRC staff requested this information to compare DR analytical results on the basis of the power-to-flow ratio. Therefore, the NRC staff finds that the response is acceptable.

The NRC staff used the flow rate and the other information provided in the response to RAI 6-12 to plot trends with measured and predicted decay ratios as a function of key reactor parameters. These parameters are, namely, the: core flow rate, core thermal power, and nodal power peaking factor. The results confirm that gross trends in the measured and predicted decay ratios are consistent with the NRC staff's expectations. The NRC staff notes **[**

]

The NRC staff also compared trends in the ratio of the predicted-to-measured DR with a parameter calculated by the NRC staff. This parameter is the product of the power and nodal peaking factor divided by the core flow rate. Trend lines in Figures A.6.1 and A.6.2 have been included for the **[**

]

The NRC staff confirmed that the POLCA-T methods are likely to **[**
] The results provided in Figure A.6.3 illustrate that POLCA-T has a **[**
] This plot demonstrates that the POLCA-T methodology is **[**

] The NRC staff notes that the **[**

]

Therefore, the NRC staff finds that the information provided was sufficient for the NRC staff to complete its assessment of the stability qualification and POLCA-T performance. The NRC staff is reasonably assured that the application of the POLCA-T methodology for the
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References

6-35.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up Response to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-36, August 22, 2008. (ADAMS Accession No. ML082520770)

6-35.2 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "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) (Proprietary/Non-proprietary)," LTR-NRC-07-53, October 9, 2007. (ADAMS Accession No. ML072900261)

RAI 6-36

The NRC staff requested additional information regarding the nodalization of the lower plenum. In particular, the NRC staff requested this information to assess the capability of POLCA-T to model the coolant temperature distribution at the core inlet under conditions where feedwater temperature reduction may lead to unstable core behavior following a dual recirculation pump trip (2RPT) dual recirculation pump trip for jet pump plants. The NRC staff was concerned that POLCA-T may not model the stratification of the flow in the lower plenum, and therefore, POLCA-T may miscalculate the core inlet enthalpy distribution in transient analyses.

The response provided in Reference 6-36.1 states that the current scope of application of POLCA-T is for steady state evaluations. **[**

]

The response further provides the results of calculations that demonstrate the sensitivity of the core decay ratio to core inlet temperature distribution. To perform this calculation, a POLCA-T core model was modified to lower the temperature of the coolant flow into one-eighth of the core bundles. The results show that the behavior is more unstable when the core inlet temperature is unevenly distributed.

Based on the results of the sensitivity analysis, the **[**

]

The NRC staff intends to **[**

]

Reference

6-36.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Third Set of Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: Systems Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-09-14, Rev. 1, March 2, 2010. (ADAMS Accession No. ML100830375)

RAI 6-37

The NRC staff requested additional information regarding the sensitivity of the stability analysis results to the gas gap properties. In particular, the NRC staff requested that Westinghouse evaluate the sensitivity of the decay ratio to the various gas gap properties. In cases where the analysis results were shown to be very sensitive, the NRC staff requested that Westinghouse provide additional information regarding the high burnup qualification of the STAV models in POLCA-T.

The response provided in Reference 6-37.1 provides the results of sensitivity studies performed using the **[** **]** The results were summarized in a table that shows that the primary variable affecting the stability calculations is **[**

]

The response states that the **[** **]** primarily influence reactor stability. The NRC staff agrees with this assessment. However, the NRC staff notes that the response has demonstrated that **[**

]

On the basis that the **[** **]** and that the parameter with the greatest influence on stability performance is accurately predicted for high exposure, the NRC staff finds that the response adequately justifies the applicability of the POLCA-T gas gap models for stability analyses of high burnup fuel.

Reference

6-37.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Third Set of Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: Systems Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-09-14, Rev. 1, March 2, 2010. (ADAMS Accession No. ML100830375)

RAI 7: Control Rod Drop Accident

RAI 7-1

The NRC staff requested that Westinghouse provide additional information regarding the adequacy of the PHOENIX4/POLCA7 methodology to calculate the plutonium contribution to the Doppler worth for CRDA analyses. The response to RAI 7-1 contains **[**

]

The Pu-238 isotope is a minor contributor to the transient response, and the NRC staff does not consider large errors in the prediction of this isotope to be consequential to accurate modeling of BWR kinetic behavior.

In addition to comparison against the Organisation for Economic Cooperation and Development / Nuclear Energy Agency (OECD/NEA) benchmark, Westinghouse provided comparisons between the PHOENIX4 isotopic predictions and HELIOS. The comparisons were performed for **[**

]

Aside from the concentration, the NRC staff is primarily concerned with the acceptable calculation of the Doppler worth. Westinghouse provided comparisons between PHOENIX and HELIOS considering the effect of fuel temperature. The cases analyzed were at **[**

]

Finally, a comparison was performed using Monte Carlo N Particle (MCNP) with PHOENIX-generated isotopics. The basis of the comparison was **[**

]

These comparison studies provide the NRC staff with reasonable assurance that the PHOENIX code is sufficiently accurate in its prediction of plutonium worth to justify the use of the code for CRDAs for modern, aggressive BWR operating strategies and modern fuel designs. Therefore, the NRC staff finds the response to be acceptable.

Reference

7-1.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

RAI 7-2

The NRC staff requested additional information regarding the xenon conditions for the initialization of the CRDA. The response to RAI 7-2 states that **[**

]

References

7-2.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Further Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-08-59, December 16, 2008.

7-2.2 CENPD-284-P-A, "Control Rod Drop Accident Analysis for Boiling Water Reactors: Summary and Qualification," ABB CE, July 1996.

RAI 7-3

At end of cycle (EOC) conditions, modern BWR cores have a positive moderator temperature coefficient. The NRC staff requested that Westinghouse provide additional details regarding the POLCA-T method in terms of accounting for this phenomenon. The response states that the **[**

] Therefore, the NRC staff finds that the response is acceptable.

Reference

7-3.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

RAI 7-4

The NRC staff requested additional information regarding the delayed neutron fraction sensitivity. The results of the delayed neutron fraction sensitivity study in the TR indicated that the CRDA analysis results were not sensitive to the delayed neutron fraction. The NRC staff notes that the delayed neutron fraction is a surrogate measure of the core power response to changes in reactivity, and the NRC staff expects the results to be sensitive to the delayed neutron fraction over the range of variation considered in the TR.

The NRC staff requested additional information regarding these analyses and cites the RAMONA-3B sensitivity analyses presented in CENPD-284-P-A as well as studies performed by Brookhaven National Laboratory (BNL) presented in BNL-NUREG-66230 and BNL-NUREG-67430.

The response states that there was an incorrect implementation in POLCA-T in the delayed neutron fraction multiplier. The incorrect implementation results in the multiplier essentially being removed after the first iteration. The response states that the incorrect implementation was corrected. The CRDA sensitivity analyses were reanalyzed using the corrected code and the results presented in the RAI response.

The response states that the affected sections of the TR will be revised in the next revision to incorporate the corrected numerical analyses and to update the uncertainty analysis.

The response provides the sensitivity analysis in tabular and pictorial form. **[**

]

The NRC staff RAI also requested that the POLCA-T results be compared against the BNL studies. The sensitivity was compared against the BNL simplistic models, **[**

]

The NRC staff finds that the unexpected results provided in the original TR were a result of an incorrect implementation of the delayed neutron fraction multiplier. Based on its review of the corrected results, the NRC staff finds that the delayed neutron sensitivity is consistent with expectations based on similar approved codes (RAMONA-3B) and simplistic physical models (BNL studies). Therefore, the NRC staff finds that the code was appropriately corrected and that the code output is acceptable for evaluating the sensitivity.

The response provides **[**

]

[

]

Therefore, the NRC staff finds that the response is adequate, and the proposed means for accounting for the delayed neutron fraction uncertainty is acceptable.

References

7-4.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up Response to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-36, August 22, 2008. (ADAMS Accession No. ML082520770)

7-4.2 CENPD-284-P-A, "Control Rod Drop Accident Analysis for Boiling Water Reactors: Summary and Qualification," ABB CE, July 1996.

7-4.3 Diamond, D., et al., "Estimating the Uncertainty in Reactivity Accident Neutronic Calculations," BNL-NUREG-66230.

7-4.4 Diamond, D., et al., "A Qualitative Approach to Uncertainty Analysis for the PWR Rod Ejection Accident," BNL-NUREG-67430.

RAI 7-5

The NRC staff requested that Westinghouse provide additional details regarding the selection of limiting control blade screening for CRDA analyses considering the effects of cycle exposure. The response states that the process outlined in Section A.4.6 specifies that the

[

] The NRC staff finds that this approach is acceptable and adequately resolves the NRC staff's concern.

Reference

7-5.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

RAI 7-6

The NRC staff requested that Westinghouse justify the control blade worth uncertainty used in the PIRT and subsequent uncertainty analysis for the CRDA analysis. The RAI specifically requested that Westinghouse use data from local critical tests to quantify the control blade worth uncertainty. The response provided cold critical test data collected over several cycles. The cold eigenvalue was used to infer the control blade worth uncertainty. In the analysis, Westinghouse determined that the POLCA7 calculated blade worth has **[**

] This uncertainty is significantly smaller than the uncertainty used in the statistical analysis of 5 percent. Therefore, the NRC staff finds that the response provides adequate reasonable assurance that the control blade worth used in the uncertainty analysis is a **[**

] and is therefore acceptable.

Reference

7-6.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

RAI 7-7

The NRC staff requested that Westinghouse provide additional information regarding the assumptions in the modeling of the reactor SCRAM. The response states that the SCRAM speeds are based on the Technical Specifications requirements for SCRAM speed. The NRC staff finds that this approach is acceptable.

Reference

7-7.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

RAI 7-8

The NRC staff requested additional information regarding the use of the ENDF/B-VI delayed neutron libraries. The response states that Westinghouse is aware that the delayed neutron libraries in ENDF/B-VI are in error. Therefore, these **[**

] The response compares the POLCA-T delayed neutron library to the RAMONA-3B library (which was based on the PHOENIX2 library). In general, these **[**

] the NRC staff finds that the delayed neutron library data incorporated in POLCA-T is acceptable for CRDA analyses.

Reference

7-8.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

RAI 7-9

The NRC staff requested additional information regarding the adequacy of the time step size and time step size control algorithms for CRDA analyses. The NRC staff particularly requested that Westinghouse justify the relative number of thermal-hydraulic and nuclear

iterations and provide adequate justification of the time step resolution to evaluate CRDA events.

The response states the **[**
] The NRC staff finds that this approach is acceptable to **[**

]

The response provides details of a sensitivity study performed for reduced time steps. **[**

] However, as depicted in Figure 2 of the response, **[**

] The NRC staff reviewed the performance of the time step control algorithm presented in Figure 5 of the response.

Figure 5 depicts the time step **[** **]** The cases considered with time steps **[**

]

The NRC staff notes that while the **[**

] Therefore, the NRC staff is not reasonably assured that the sensitivity study provides an adequate basis that **[**

]

The NRC staff issued a supplemental request for additional information to RAI 7-9S1. The NRC staff requested that Westinghouse consider smaller time step sizes to ensure that the **[** **]** and that the selection of the maximum time step size for CRDA analysis is acceptably small.

Supplemental Information Provided in Response to RAI 7-9S1

Westinghouse provided supplemental information to RAI 7-9 in Reference 7-9.2. The response provides the results of additional sensitivity studies for smaller time steps. Figure 5a of the response demonstrates that the additional time step sizes **[**

] Therefore, the NRC staff finds that these analyses consider the appropriate range of time step sizes to test convergence. Figure 2a of the response shows the sensitivity of the power pulse peak and timing of the peak power to the maximum time step size. Figure 2a shows that the **[**

] The results provided in Table 1a for all of the time step sizes considered demonstrate **[** **]**

The response further states that the time step upper limit is **[**

]

The response states that **[**

]

On the basis of the relative insensitivity of the figure of merit (fuel enthalpy), the NRC staff finds that this approach is sufficient to ensure adequate time step resolution and stability of the solution for CRDA analyses.

References

7-9.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up Response to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-36, August 22, 2008. (ADAMS Accession No. ML082520770)

7-9.2 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Further Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-08-59, December 16, 2008. (ADAMS Accession No. ML083660101)

RAI 7-10

Due to the highly decoupled nature of the reactor during conditions typical of CRDA analysis initial conditions, the NRC staff requested additional qualification of the POLCA-T pin power reconstruction model. The bundle power is expected to be highly radially peaked during the power excursion, and accurate modeling of the flux shape during the CRDA is required to accurately predict the hot rod integral power.

The response to RAI 7-10 (Reference 7-10.1) provides comparative analyses performed with POLCA-T and PHOENIX4. PHOENIX4 is a detailed two-dimensional lattice physics code that **[**

]

The comparative analyses considered six core configurations. These configurations include variation in the **[**

]

The NRC staff finds that the comparative analyses provided are sufficient to justify the accuracy of the POLCA-T pin power reconstruction model for use in CRDA analyses.

References

7-10.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Further Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-08-59, December 16, 2008.

7-10.2 CENPD-390-P-A, "The Advanced PHOENIX and POLCA codes for Nuclear Design of Boiling Water Reactors" ABB CE, December 2000. (ADAMS Accession No. ML010100268)

RAI 7-11

The NRC staff requested additional information on whether the model capability in RAMONA-3B to analyze off-center control rods as effective central control rods was maintained

in POLCA-T. The response states that **[** **]** (Reference 7-11.1). The NRC staff finds that this approach would yield more accurate results and is therefore acceptable.

Reference

7-11.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

RAI 7-12

The NRC staff requested additional information regarding conservatism in the hot rod gas gap conductance input or modeling. The response states that POLCA-T considers **[**

] (Reference 7-12.1).

The NRC staff requested additional information since the sensitivity of the CRDA analysis results to the gap conductance is not straightforward. The NRC staff agrees with the statements made in the response describing the competing effects in terms of fuel temperature and negative Doppler worth having a compensating effect on the analysis. The response, however, states that performed sensitivity studies demonstrate **[**

]

CENPD-284-P-A considered the sensitivity of heat transfer models in the RAMONA-3B methodology for CRDA analysis. Section 6 of Part III of CENPD-284-P-A contains sensitivity studies performed with RAMONA-3B. **[**

] (References 7-12.2 and 7-12.3).

Supplemental Information Provided for Response to RAI 7-12

In Reference 7-12.5, Westinghouse provided supplemental information to the response to RAI 7-12. The original response quoted sensitivity studies that were not provided in the original response. The supplemental information provides the results of sensitivity studies performed using RAMONA-3 over a wide range of gas gap heat transfer coefficients. This study illustrates the variation in fuel enthalpy as the gas gap heat transfer coefficient is varied from very low values to very high values. **[**

] These studies augment the original bounding analysis in Reference 7-12.4.

On the basis that **[**

] the NRC staff concludes that the gas gap heat transfer coefficient is conservatively modeled in POLCA-T for CRDA analyses. Therefore, the NRC staff finds that the response and analytic approach are acceptable.

References

7-12.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Further Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-08-59, December 16, 2008.

7-12.2 CENPD-285-P-A, "Fuel Rod Design Methods for Boiling Water Reactors," ABB CE, July 1996.

7-12.3 WCAP-15836-P-A, "Fuel Rod Design Methods for Boiling Water Reactors – Supplement 1," Westinghouse Electric Company, April 2006.

7-12.4 CENPD-284-P-A, "Control Rod Drop Accident Analysis for Boiling Water Reactors: Summary and Qualification," ABB CE, July 1996.

7-12.5 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Third Set of Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: Systems Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-09-14, Rev. 1, March 2, 2010. (ADAMS Accession No. ML100830375)

<u>RAI 7-13</u>

In RAI 7-13 the NRC staff requested additional information regarding the input assumptions. The response states that the cold clean initial conditions are the limiting conditions for CRDA analyses (Reference 7-13.1). The response refers to Section 4 of CENPD-284-P-A. The CENPD-284-P-A TR (Reference 7-13.2) states:

[

]

The response to RAI 7-13 states, however, that the cold clean initial condition is assumed to

provide for an unambiguous initial condition. The response references Figure 4.5.14 of Reference 7-13.2. The figure shows **[**

]

On the basis of these arguments and the sensitivity analyses provided in Figure 4.5.14 of Reference 7-13.2, the NRC staff finds that the cold clean initial conditions are appropriate for licensing basis CRDA analyses using POLCA-T.

References

7-13.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Further Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-08-59, December 16, 2008.

7-13.2 CENPD-284-P-A, "Control Rod Drop Accident Analysis for Boiling Water Reactors: Summary and Qualification," ABB CE, July 1996.

RAI 7-14

The NRC staff requested additional information regarding the prediction of temperature/reactivity feedback in the POLCA-T transient evaluation of CRDA events. The NRC staff specifically requested additional information regarding the reactivity effect of the changing pellet dimensions during CRDAs.

During a CRDA, the power excursion is terminated by compensating Doppler reactivity worth due to increasing fuel temperature. The increased fuel temperature, however, results in changes in the pellet geometry. In particular, the pellet size will increase due to thermal expansion, and the pellet density will decrease. This does not have a clear effect on the reactivity as the resonance absorption is expected to trend as the square root of the surface area to the mass of the heavy metal in the node.

Additionally, the NRC staff requested information regarding volatile nuclides. The NRC staff requested that Westinghouse consider the case where elevated fuel temperatures result in the release of highly absorbing volatile fission products, thus potentially resulting in an increase in nodal reactivity due to the escape of absorbing material in the pin plena.

The first part of the response considers the **[**

]

A simplified Nordheim-Fuchs model was used **[**

]

Results indicate that for **[**

] As stated previously expansion will increase the pellet surface, but will also reduce the density. **[**

]

The response states that the **[**

] The NRC staff finds that the analyses provide an acceptable basis for the **[** **]** provided in the response.

The NRC staff has **[**

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Reference

7-14.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up Response to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-36, August 22, 2008. (ADAMS No. ML082520770)

RAI 7-15

The NRC staff requested additional information about the assumptions regarding operator error in the CRDA analysis process. The response states that the **[ ]** are explicitly accounted for in the analysis. These assumptions are **[ ]** The methodology quoted in the response relies on **[ ]** This approach is fully consistent with the approach approved by the NRC staff for the RAMONA-3B methodology for CRDA analysis (Reference 7-15.2). The NRC staff finds that the proposed TR revision and the description of the methodology in the RAI response are adequate and acceptable.

References

7-15.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

7-15.2 CENPD-284-P-A, "Control Rod Drop Accident Analysis for Boiling Water Reactors: Summary and Qualification," ABB CE, July 1996.

RAI 7-16

The NRC staff requested clarification of the terms "power" and "flux" SCRAM. The response provided by Westinghouse states that the **[ ]** The "flux" SCRAM refers to the reactor SCRAM initiated by an APRM reading of 120 percent neutron flux. The response states that SCRAM delay times are conservatively determined and included in the analysis. The response also **[ ]** The NRC staff finds that the response and TR revision are acceptable and provide adequate clarification.

Reference

7-16.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

RAI 7-17

The NRC staff requested additional clarification of the results presented in Table A.3-6. In response to the NRC staff's question, Westinghouse provided a revised table that more clearly defines the parameters presented in the table. Specifically, the table clarifies the integrated power and the time to which the power is integrated. The results quote the POLCA-T calculated integrated power and the POLCA-T calculated power integrated up to the time at which peak power was measured in the SPERT test.

Reference

7-17.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

RAI 7-18

The NRC staff requested additional information regarding the POLCA-T SPERT III E qualification presented in Appendix A of the TR. First, the NRC staff requested the power shape qualification data. Second, the NRC staff requested the comparison of POLCA-T against the case 18 test. Third, the NRC staff requested that Westinghouse provide a figure depicting the Doppler worth sensitivity.

First, the response states that SPERT III E power shapes were not measured and that the TR language was referring to the shapes of the transient power histories (Reference 7-18.1). The explanation is sufficient to clarify the TR description. The NRC staff requested separately in RAI 7-10 that Westinghouse qualify the POLCA-T pin power reconstruction model. The response to RAI 7-10 provides an adequate basis for the NRC staff to accept the power distribution modeling capability of POLCA-T.

Second, the response provides a figure showing the comparison of POLCA-T to the case 18 SPERT III E test (Reference 7-18.1). Figure A.3-10a in the response and Figure A.3-10 from the TR indicate that POLCA-T **[**

]

Third, the response provides a figure similar to Figure 5.3.16 of CENPD-284-P-A (Reference 7-18.1). Figure 5.3.16 and Figure 5.3.17 of CENPD-284-P-A illustrate **[**
] (Reference 7-18.2). Only SPERT III E case 43 is considered in Figure A.3-10b of the response. When compared with Figure 5.3.17 of CENPD-284-P-A, the response **[**

]

This provides the NRC staff with assurance that the code system predicts sensitivities that are consistent with the NRC staff's expectations based on the PIRT and previous sensitivity studies.

Therefore, the NRC staff finds that the response is acceptable.

References

7-18.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Further Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-08-59, December 16, 2008. (ADAMS Accession No. ML083660101)

7-18.2 CENPD-284-P-A, "Control Rod Drop Accident Analysis for Boiling Water Reactors: Summary and Qualification," ABB CE, July 1996

RAI 7-19

The NRC staff requested additional information regarding the sensitivity of the CRDA analyses to the core flow rate. The response provides Table A.5-8a (Reference 7-19.1). The results provided in the table demonstrate that **[**

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The NRC staff finds that the results are acceptable to address the NRC staff's concerns regarding this code sensitivity. The treatment of the core mass flow rate and associated uncertainty are therefore acceptable since the **[**
]

Reference

7-19.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Further Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-08-59, December 16, 2008. (ADAMS Accession No. ML083660101)

RAI 7-20

The NRC staff requested that Westinghouse clarify if the POLCA-T screening criteria were revised relative to the RAMONA-3B screening criteria for dynamic CRDA analysis. The response states that the POLCA-T criteria **[** **]** to the previously approved RAMONA-3B screening criteria. The NRC staff has previously reviewed these criteria and found them acceptable. Therefore, the NRC staff finds that the approach and the response to the RAI are acceptable.

References

7-20.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

7-20.2 CENPD-284-P-A, “Control Rod Drop Accident Analysis for Boiling Water Reactors: Summary and Qualification,” ABB CE, July 1996.

RAI 7-21

The NRC staff requested additional information regarding the PB2 TT test qualification presented in Appendix A of the TR.

The NRC staff requested additional information regarding the calculated axial power profiles in Figures A.3-4 and A.3-5. The response states that the “PHOENIX4 XS data” plots are generated by using upstream predictive cycle follow calculations using POLCA7/PHOENIX4 codes. The “PSU XS data” cases are presented for information only. “The PSU XS data” power profile is determined using a POLCA-T calculation with cross section data that was generated external to the Westinghouse process.

The NRC staff requested additional information regarding what the terms P1, APRM Probe 1 and APRM Probe 2 referred to. The NRC staff reviewed the information provided in the response and Reference 7-21.2. P1 refers to the process computer calculation of the PB2 EOC2 axial power shape. APRM probes refer to the APRM Channel A power measurement and the 80 LPRM normalized power measurement. The response likewise clarifies, as is described in Reference 7-21.2 that the five TIP instruments were inserted in the core during the TT tests.

The NRC staff requested that the term “m/sec” be revised to more clearly indicate milliseconds. The response provides the revised table. The NRC staff also requested that the term “measured” be clarified. The revised table provides a footnote discussing the term by stating that the measured value is based on the average of 80 LPRM probe responses.

The NRC staff found that the clarification provided in the response was sufficient for the NRC staff to complete its review of the subject matter. Therefore, the response was acceptable.

References

7-21.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, “Further Responses to the Second Round of NRC’s Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, ‘POLCA-T: System Analysis Code with Three-Dimensional Core Model’ (TAC No. MD5258) (Proprietary/Non-Proprietary),” LTR-NRC-08-59, December 16, 2008. (ADAMS Accession No. ML083660101)

7-21.2 EPRI NP-564, “Transient and Stability Tests at Peach Bottom Atomic Power Station Unit 2 End of Cycle 2,” Electric Power Research Institute, June 1978.

RAI 7-22

In RAI 7-22, the NRC staff requested several figures substantially similar to the figures provided in the TR labeled Figures A.3-6 through A.3-9. The NRC staff requested that Westinghouse provide figures that are similar to these figures with the power response shifted on the x-axis so that the time scales are identical starting at the point of the initial core pressure response. This allows the NRC staff to compare the dynamic reactivity prediction by POLCA-T to the power response normalized by the back pressure wave transit time from the turbine stop valves to the core. The measured and predicted core pressure response times were **[**

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The plots include the predicted and measured total neutron powers as well as individual LPRM predictions and measurements. Figures A.3-6a and A.3-7a **[** **]** Generally, the peak power and the integral power are **[** **]** Relative to the data, POLCA-T shows a **[** **]** The timing difference in terms of the peak power is approximately **[** **]** TT tests. The overall shape of the transient response is **[** **]** demonstrating that the POLCA-T code is accurately predicting the void reactivity response to the pressurization.

The LPRM responses are **[** **]** to the overall core power responses. This demonstrates that the POLCA-T code is predicting the local transient power correctly. Some of the LPRM responses **[**

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On the basis of these revised plots, the NRC staff finds that POLCA-T accurately predicts the peak power, the integral power, and the character of the transient power response. The NRC staff has found **[** **]** The transient power shape is also accurately modeled. For the subject analyses (CRDA and stability), the NRC staff finds that the important phenomena are accurately modeled and that the figures of merit are acceptably predicted.

The comparison of the POLCA-T calculation to the PSU XS data calculation indicates **[**

]

The **[**

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Reference

7-22.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Further Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-08-59, December 16, 2008. (ADAMS Accession No. ML083660101)

<u>RAI 7-23</u>

The NRC staff requested that Westinghouse provide the results of the simulation of the third PB2 EOC2 TT test. The response states that only the steady state simulation was performed. The transient response calculations will be provided to the NRC in the POLCA-T for transient application (Appendix C). The response states that the text in the TR will be revised. The NRC staff finds that this correction is acceptable. The NRC staff has reviewed the qualification data provided in the Appendix A of the TR and found that there are sufficient qualification cases in the TR for the NRC staff to reach a reasonable assurance finding regarding the applicability of POLCA-T to CRDA analyses. Therefore, the NRC staff finds that the response is adequate.

Reference

7-23.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

RAI 7-24

The NRC staff requested additional information regarding the axial power shape uncertainties. The NRC staff requested that Westinghouse compute the axial power shape root mean square (RMS) differences based on the P1 edit for the PB2 TT tests. The response provides the comparisons that demonstrate that the POLCA-T-predicted power shapes are **[ ]** with the P1 edit. The response similarly provides the RMS differences for the PSU XS data. The resultant RMS differences using the Westinghouse calculation process indicate that the uncertainties are **[ ]** with the previously-established uncertainties in CENPD-390-P-A.

The table in the response quotes a RMS error **[ ]** according to CENPD-390-P-A). However, Westinghouse notes that the **[ ]**

Westinghouse provided **[ ]** in CENPD-390-P-A.

Table A.3-3b of the response to RAI 7-24 provides the RMS differences of **[ ]** These values are **[ ]** Therefore, the NRC staff finds that the **[ ]**

Reference

7-24.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Further Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-08-59, December 16, 2008. (ADAMS Accession No. ML083660101)

RAI 7-25

In the event that a CRDA analysis is not bound by a previous analysis, the NRC staff requested that Westinghouse provide the details for how the radiological consequences are determined and evaluated against acceptance criteria. The response states that if radiological consequences must be evaluated, these consequences will not be evaluated using POLCA-T.

The radiological consequences will be evaluated using either Regulatory Guide (RG) 1.183 (alternate source term) or RG 1.195 (traditional method). The NRC staff finds that this approach is acceptable.

When determining the radiological fission product inventory, the traditional (RG 1.195) and alternative (RG 1.183) source methods must include an increased inventory to account for transient fission gas release for new reactor applications. Therefore, the NRC staff imposes the condition for ABWR CRDA analyses and dose assessment that the transient FGR must be calculated according to the following correlation from Appendix B of Section 4.2, "Review of Transient and Accident Analysis Methods", of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP 4.2):

Transient FGR = {(0.2286*ΔH) – 7.1419}
Where:
FGR = Fission gas release, % (must be > 0)
ΔH = Increase in fuel enthalpy, Δcal/g

The transient release from each axial node which experiences the power pulse may be calculated separately and combined to yield the total transient FGR for a particular fuel rod. The combined steady state gap inventory and transient FGR from every fuel rod predicted to experience cladding failure (all failure mechanisms) should be used in the dose assessment.

References

7-25.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

7-25.2 Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2008. (ADAMS Accession No. ML003716792)

7-25.3 Regulatory Guide 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," May 2004. (ADAMS Accession No. ML031490640)

RAI 7-26

The NRC staff requested additional information regarding the use of the interim acceptance criteria in SRP Section 4.2 Revision 3. These interim criteria are provided for reactivity insertion accidents for new plant applications. Specifically, the NRC staff requested that the TR be revised to capture these interim acceptance criteria for new plant applications, and that Westinghouse provide additional details regarding the calculation of the rod internal rod pressure as this pressure is used to determine compliance with the interim criteria.

The response provided in Reference 7-26.1 describes how the interim acceptance criteria are used and that the TR will be revised to incorporate the interim acceptance criteria for new plant applications. The NRC staff finds that this is acceptable.

The response also provides details of the POLCA-T calculation of the rod internal pressure. The calculation is performed **[**

] The NRC staff finds that this approach is acceptable to calculate the number of damaged fuel rods.

Therefore, the NRC staff finds that the use of the interim acceptance criteria for new plant applications has been adequately incorporated and the calculational methodology is sufficiently conservative to evaluate the number of damaged fuel rods. On these bases, the response is acceptable.

Reference

7-26.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Fourth Set of Responses to the Second Round of NRC's Request of Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-09-21, April 8, 2009. (ADAMS Accession No. ML100281005)

RAI 8: Thermal Hydraulics

RAI 8-1

The NRC staff requested additional information regarding the dryout correlation library. In particular, the NRC staff requested that Westinghouse specify the correlations in the library, the applicable fuel design, and reference to the experimental data used to develop the correlation. The response provides the requested information for Westinghouse fuel designs currently operated in the U.S. These include the applicable correlations for SVEA-96, SVEA-96+, and SVEA-96 Optima2.

The response states that internal Westinghouse requirements assure that the use of NRC approved correlations for licensing analyses specify the correlation used, refer to the NRC approved documentation, and explain how the correlation is used within the approval.

The NRC staff finds that the response is sufficient in specifying how Westinghouse treats critical power evaluations for Westinghouse fuel designs. However, the response does not provide details regarding the mixed core application.

The subject review of POLCA-T to stability and CRDA analyses, however, does not require evaluation of the critical power ratio. The NRC staff notes that the stability TR Appendix B does not seek approval for POLCA-T to develop DIVOM slopes. Therefore, for the subject review, the calculation of the critical power ratio is ancillary to the analyses.

The NRC staff, therefore, defers the review of the subject of the applicability of the dryout correlations to review of POLCA-T for application to either transients or to generate a DIVOM curve. In these subsequent reviews, the NRC staff will address the application of POLCA-T to mixed core evaluations.

References

8-1.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up Response to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-36, August 22, 2008. (ADAMS Accession No. ML0825207705)

RAI 8-2

The NRC staff requested additional information regarding the H1 and H6 heat transfer coefficients. The NRC staff notes that these **[**

] The NRC staff finds that this is acceptable.

Reference

8-2.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

RAI 8-3

The NRC staff requested additional information regarding the heat transfer regime map. The map provided in the original submittal of the subject TR does not include consideration of the Reynolds number. The figure, which is provided in the response, indicates that in particular regimes, a series of correlations is available to predict the heat transfer. While the NRC staff notes that full scale qualification of the heat transfer predictive capabilities of POCLA-T was provided in the response to RAI 3-5, the NRC staff is not aware how interpolations are performed or how particular correlations are selected to (1) ensure that the phenomena are modeled accurately, and (2) that there are no potential conditions where discontinuity in the heat transfer predictions results in inaccuracy or numerical instability.

More detailed maps are provided in the response for the non-dryout and post-dryout heat transfer regimes. The figures illustrate the transitions between the different correlations based on the transition Reynolds numbers. Following the figures, the response provides the application range for each correlation, and the maps specify where interpolation is performed. The NRC staff finds that the details provided are sufficient for the NRC staff to understand how POLCA-T selects the correlation and evaluates the heat transfer coefficient for **[** **]** In each case, the NRC staff reviewed the response and found that the interpolations and use of each correlation allow for the accurate prediction of the coefficient and that these are adequately interpolated to preclude discontinuity.

As previously stated, the qualification of POLCA-T to evaluate heat transfer characteristics is provided separately under RAI 3-5.

Reference

8-3.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

RAI 8-4

The NRC staff notes an error in the POLCA-T model for Countercurrent Flow Limit (CCFL) for modern fuel bundle geometries. In its response to this RAI, Westinghouse provides a commitment to update the model and provide this revision to the NRC with the Appendix D POLCA-T application to ATWS. As countercurrent flow is not expected to occur for CRDA or during damped thermal- hydraulic density wave oscillation events, the NRC staff finds that the response is sufficient for the NRC staff to complete its review of the subject TR. The NRC staff will impose a condition that the CCFL correlation be revised to be consistent with the model submitted to address potential non-conservatisms for SVEA-96 Optima2 prior to POLCA-T's application to transient analyses where countercurrent flow may occur.

Reference

8-4.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

RAI 8-5

The NRC staff requested additional information regarding the qualification of the void quality correlations to higher pressures and higher void fractions. In particular, the NRC staff requested that Westinghouse justify the application of the correlation to these higher pressures and voids that may be encountered under transient or accident conditions. In particular events such as main steam isolation valve (MSIV) closure without position SCRAM may result in high pressures and high void fractions.

The response is based on comparisons to other void quality correlations. In the review of WCAP-16606-P-A, the NRC staff reviewed the application of the AA78 correlation in BISON to simulate transient thermal-hydraulic conditions at these higher void and higher pressure conditions typical of ATWS scenarios. As ATWS evaluations consider vessel pressurization without SCRAM and the subsequent recirculation pump trip, these events constitute a reasonable basis to establish the highest pressures and void fractions for which the correlations are used.

In its review of WCAP-16606-P-A, Westinghouse describes a methodology for using the EPRI void quality correlation (Chexal-Lellouche) to **[**

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The response includes detailed comparisons between the Chexal-Lellouche and the DF01 and DF02 correlations for various void fractions and pressures. The response indicates
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Supplemental Information Provided in Response to RAI 8-5S1

The NRC staff noted an error in the units in Table 3. The NRC staff communicated this typographical error in the form of an RAI requesting that the error be corrected in the approved revision of the TR. In Reference 8-5.3, Westinghouse acknowledges that a decimal point is missing in the table. The response states that the table will be corrected accordingly in the approved TR version. The NRC staff finds that this is acceptable.

References

8-5.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up Response to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-36, August 22, 2008. (ADAMS Accession No. ML082520770)

8-5.2 WCAP-16606-P-A, "Supplement 2 to BISON Topical Report RPA 90-90-P-A," Westinghouse Electric Company, January 2008. (ADAMS Accession No. ML081280718)

8-5.3 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Further Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-08-59, December 16, 2008. (ADAMS Accession No. ML083660101)

<u>RAI 8-6</u>

The NRC staff requested additional information regarding the momentum equation. The NRC staff had several questions regarding its application for several features common in reactor simulation models, including elbows, tees, and horizontal flows.

(a) Flow directions that are not vertical

The response states that for flow directions that are horizontal, the gravity term is eliminated from the momentum equation. For inclined flow paths, the gravity term is weighted according to the cosine of the angle of the incline. The NRC staff finds that this is acceptable.

(b) Flow through an elbow

The response states that the user must specify the elbow loss coefficient and the surface roughness. In cases where the elbow is represented by several volume cells, the loss coefficient is distributed through the multiple cells equally. The NRC staff finds that this is acceptable.

(c) Multiple neighboring cells

The response provides a discussion for the flow distribution in the case of tee junctions. For these junctions, a user-supplied factor is used to multiply the flux pressure drop to transport the appropriate amount of momentum through the downstream flow path. The NRC staff audited the implementation of the momentum equation in the POLCA-T source code for tee junctions and found that the implementation was acceptable (see Section 5.3 of Reference 8-6.1). Therefore, the NRC staff finds that this is acceptable.

(d) Plena

The response states that for plena, the flow junction at each parallel flow path references the common manifold cell, and the solution of the equation explicitly accounts for the differences in the upstream and downstream cells. The NRC staff finds this acceptable.

(e) Virtual mass

The NRC staff requested that Westinghouse describe how the virtual mass effect is captured. The response states that the **[** **]** The NRC staff agrees and finds this acceptable.

(f) Two-phase cell parameters

The NRC staff requested that Westinghouse rewrite the momentum equation to demonstrate how the equation is solved based on the fluid cell parameters. The response states that the **[** **]** The single fluid properties for the calculation come from the neighboring volume cells, and a single set of linear equations is developed for the entire system and is solved during each iteration. The NRC staff finds this acceptable.

(g) Interfacial shear

The response states that interfacial shear is not explicitly accounted for. The POLCA-T fluid model is a **[** **]** The void quality correlation captures the effects of interfacial shear implicitly. The NRC staff agrees and finds this acceptable.

(h) and (i) Countercurrent flow

The response states that under conditions of countercurrent flow, the **[** **]** The NRC staff finds this approach acceptable.

(j) Wetted perimeter

The response states that the calculation of the wetted perimeter is **[** **]**

[

] The NRC staff finds that this approach is reasonable.

(k) Velocity distribution correction factor

The NRC staff requested that Westinghouse provide the basis for the velocity distribution correction factor. The response states that the distribution factor accounts for the velocity distribution in the channel when voiding occurs. The **[**

] It is intended to account for the effect of the two-phase flow on the friction on the velocity distribution near the wall. The correction factors are determined based on full-scale bundle pressure drop data.

(l) Sudden pressure drop

The NRC staff requested that Westinghouse evaluate the performance of the single-fluid momentum equation where sudden pressure drops result in downstream void formation. In response to the RAI, Westinghouse provided qualification of the POLCA-T model to the Edwards experiment. The Edwards pipe test is a rapid blow-down test whereby a pressurized canister is depressurized following the explosive opening of one end of a vessel. The test measured various dynamic pressures, and the initial outlet velocity is choked. The **[**

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The **[**

] Therefore, the NRC staff finds that the response provides reasonable assurance that the POLCA-T thermal-hydraulic model can simulate this phenomenon and is acceptable.

References

8-6.1 NRC Audit Results Summary Report "WCAP-16747-P: POLCA-T System Analysis Code with Three-Dimensional Core Model", May 2010. (ADAMS Accession No. ML100840695)

8-6.2 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

<u>RAI 8-7</u>

The NRC staff requested that Westinghouse provide an analysis to demonstrate the efficacy of the momentum equation by doing a sensitivity study on a complex reactor systems model. The NRC staff requested that all energy sources be turned off in a model and the model run to ensure that the solution to the momentum equation did not result in artificial momentum sources. To this end, Westinghouse provided an analysis of a **[**

] The NRC staff requested that a complex model be used to verify the solution of the momentum equation for many of the cases for which the NRC staff requested additional

information in RAI 8-6. To this end, the NRC staff issued a supplemental request for additional information. The NRC staff finds that a simple model is adequate to address the NRC staff's concerns if this model includes more complex features.

Supplemental Information Provided for Response to RAI 8-7S1

The response to RAI 8-7S1 was provided in Reference 8-7.2. The NRC staff requested analyses be performed on a sample problem to test the conservation of momentum. The test problem included consideration of flow splitting, inclined channels, and elbows. The second test problem considered the significant geometric features mentioned in the NRC staff's RAI 8-6. The results **[**

] Therefore, the response provides an acceptable and adequate basis for the NRC staff to be reasonably assured that the implementation of the momentum equation in POLCA-T does not introduce artificial momentum sources. Therefore, the NRC staff finds that the response is acceptable.

References

8-7.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

8-7.2 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Further Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-08-59, December 16, 2008. (ADAMS Accession No. ML083660101)

<u>RAI 8-8</u>

The NRC staff requested that Westinghouse provide additional details regarding the use of historical models in POLCA-T. The NRC staff's request was divided into three areas: the PARA steam line model, BISON pump model, and BISON steam separator model.

PARA Steam Line Model

The NRC staff requested that Westinghouse describe how conditions imposed on the PARA steam line model would be implemented for its use in POLCA-T. The response states that approval of the historical model is only sought so that previously developed PARA models may be used for licensing evaluations. The response states that new PARA models of the steam line will not be used in POLCA-T licensing. As the historical use of the model is requested only for use with previously developed models, the NRC staff finds that the PARA conditions need not be applied to future POLCA-T licensing calculations that are performed with explicit POLCA-T modeling of the steam line. Similarly, the NRC staff finds that the PARA models were developed consistent with the approved methodology, and therefore, have intrinsically met the conditions imposed by the NRC staff on the use of PARA. The NRC staff will impose a condition that only

PARA models that have been previously developed in accordance with the approved methodology and consistent with the NRC staff's conditions and limitations may be used with POLCA-T.

BISON Pump Model

The response states that POLCA-T does not use any of the historical BISON pump models. Therefore, the NRC staff does not require any additional information in regards to this model to complete its review.

Steam Separator Model

The NRC staff requested that Westinghouse compare the previously-approved L/A (length/area) model for the steam separator to the POLCA-T model. The response states that the POLCA-T model for the steam separator was explicitly qualified against tests performed for the ASEA-ATOM AS16 steam separator design as well as the new AS01 design developed in the late 1990's.

The qualification data was provided in the response. The NRC staff finds that the **[**

] Therefore, the NRC staff finds that the use of this model is acceptable.

Reference

8-8.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

RAI 9: Power

RAI 9-1

The NRC staff requested additional information regarding the POLCA-T calculation of the reactor power. Following a reactor SCRAM, the power generation includes sources from transient fission power (during the rod insertion and from delayed neutrons), fission product decay, actinide decay, decay of structural activation products, heat transfer from vessel internals, and exothermic energy release from metal-water reactions.

The response states that the POLCA7 neutronic code is used to calculate the fission power. The fission power is divided into two parts, that part deposited directly in the coolant (direct moderator heat) and the heat deposited in the fuel rods. The code calculates the prompt fission and delayed fission. During its audit of POLCA-T, the NRC staff reviewed the documentation

and found that the POLCA-T neutron kinetics solver is based on a **[** **]** are widely used and have previously been approved by the NRC staff in similar applications. The NRC staff finds this model to also be acceptable.

The decay power is calculated from the ANS Standard 5.1 **[** **]** is widely used for this application, and the NRC staff finds that its use is acceptable.

The stored energy is calculated according to the solution to the heat transfer and conduction equations for each heat structure included in the core model. Each thermal mass is assigned a heat structure to determine its transient variation in temperature and stored energy. The NRC staff finds, therefore, that the POLCA-T solution technique explicitly accounts for the stored energy.

The NRC staff notes that significant cladding heat-up is not expected during CRDA analyses, simulations of oscillations indicating core stability, or in the analysis of transients indicating margin to boiling transition. Therefore, the NRC staff does not require specific details of the metal-water reaction model for the subject TR review. However, the NRC staff will require this information to complete its review of the Appendix D POLCA-T application to ATWS. The response states that two options are available in POLCA-T. The first is the Baker-Just model (conservative) or the Cathcart-Pawel model (best estimate). The NRC staff defers review of these models to the review of POLCA-T for ATWS evaluations. The NRC staff will request, at a minimum, (should the NRC staff review Appendix D), that Westinghouse specify whether the Cathcart-Pawel model used in POLCA-T is the 1976 or 1977 model.

References

9-1.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, “Response to Second Round of NRC’s Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, ‘POLCA-T: System Analysis Code with Three-Dimensional Core Model’ (TAC No. MD5258) (Proprietary/Non-proprietary),” LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

9-1.2 NRC Audit Results Summary Report “WCAP-16747-P: POLCA-T System Analysis Code with Three-Dimensional Core Model”, May 2010. (ADAMS Accession No. ML100840695)

RAI 10: Control Systems

RAI 10-1

The NRC staff requested additional information regarding the control system models in POLCA-T. Control systems are modeled to change reactor system parameters to simulate the function of plant equipment and automated functions of systems. The NRC staff specifically requested information regarding the use of the control system models to model proportional integral derivative (PID) controllers.

The response states that **[**

]

On the basis of the information contained in the response, the NRC staff finds that POLCA-T contains sufficient control system model complexity to model most, if not all, control systems necessary for the accurate simulation of several transient and accident conditions. The response states that the previously submitted SAFIR control system models can be used for cases where additional control system model capability is required for specific applications.

Therefore, the NRC staff finds that the control system models are adequately described and sufficient in their capability to perform transient and accident analyses when considered with the response to RAI 10-2.

Reference

10-1.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up Response to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-36, August 22, 2008. (ADAMS Accession No. ML082520770)

RAI 10-2

The NRC staff requested additional information regarding the control system. The response to the RAI provides **[**

] The response provides an adequate basis for the NRC staff to be reasonably assured that the POLCA-T control system methodology incorporates sufficiently sophisticated methods and modeling capability to account for standard reactor control systems for transient and accident analyses when considered with the response to RAI 10-1.

Reference

10-2.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up Response to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-36, August 22, 2008. (ADAMS Accession No. ML082520770)

RAI 11: Fuel Rod Model

RAI 11-1

The NRC staff requested additional information regarding the hot rod environment. The response to RAI 11-1 states that assemblies are modeled in POLCA-T with **[**

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The NRC staff agrees with the response and finds that it is acceptable.

Reference

11-1.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

RAI 11-2

The NRC staff requested clarification of the radiation heat transfer and cladding temperature variables in POLCA-T. The response confirms that the variable T_c refers to the cladding inner surface temperature. Similarly, the response confirms that an error in the radiation heat transfer equation is a typographical error. The response provides a portion of the source code model to verify that the error does not propagate to the analyses. Therefore, the NRC staff finds that the response is acceptable.

Reference

11-2.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

RAI 11-3

The NRC staff requested additional information regarding the gas gap heat conduction model for cracked pellets. The response is adequate insofar as it provides details of the conduction model for the cracked pellet that is consistent with the model previously approved by the NRC staff in WCAP-15836-P-A.

The NRC staff requested additional details regarding the application of the models to MOX fuel. The response clarifies that approval of the model for MOX fuel is not being sought in the current application. Therefore, the use of the model is within the scope of the NRC staff's original approval.

The NRC staff requested clarification of a statement made in Section 14.2 of the TR regarding pellet cracking. The response clarifies the statement in Section 14.2 by stating that **[**

] The NRC staff finds that the **[**

]

The response also states that a statement in the TR regarding the fuel restructuring will be removed. The response states that fuel restructuring is taken into account in order to model high burnup enhanced fission gas release. This approach is consistent with the previously approved model in WCAP-15836-P-A, and, therefore, is acceptable.

References

11-3.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

11-3.2 WCAP-15836-P-A, "Fuel Rod Design Methods for Boiling Water Reactors – Supplement 1," Westinghouse Electric Company, April 2006. (ADAMS Accession No. ML061220450)

RAI 11-4

The NRC staff requested additional information regarding the thermal expansion coefficient for high gadolinia loadings. The response to RAI 11-4 states that test data confirm that the thermal expansion coefficient for uranium-dioxide-gadolinia mixtures remains consistent over a wide range of gadolinia concentrations up to **[ ]**

However, while the data indicate that the thermal expansion model may be suitable for application to gadolinia concentrations as high as **[ ]** the NRC staff did not perform a review of the applicability of POLCA-T to gadolinia concentrations above **[ ]**

The NRC staff notes that the upstream codes STAV7.2 and PHOENIX4 are only approved for application to gadolinia concentrations of **[ ]** The NRC staff will impose the condition that POLCA-T is only applicable to the analysis of cores loaded with gadolinia-bearing fuel within the minimum-approved-maximum-gadolinia-concentration of either STAV7.2 or PHOENIX4/POLCA7 as documented in WCAP-15836-P-A and CENPD-390-P-A, respectively, or in subsequently approved submittals.

References

11-4.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

11-4.2 WCAP-15836-P-A, "Fuel Rod Design Methods for Boiling-Water Reactors – Supplement 1," Westinghouse Electric Company, April 2006. (ADAMS Accession No. ML061220450)

11-4.3 CENPD-390-P-A, "The Advanced PHOENIX and POLCA codes for Nuclear Design of Boiling Water Reactors" ABB CE, December 2000 (ADAMS Accession No. ML010100268).

RAI 11-5

The NRC staff requested additional information regarding the FGR model and the effect of pellet cracking. The response states that the fission gas release is based on the approved WCAP-15836-P-A model in STAV7.2, and that the pellet cracking effect is **[**

] The NRC staff finds that the response is acceptable.

References

11-5.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

11-5.2 WCAP-15836-P-A, "Fuel Rod Design Methods for Boiling-Water Reactors – Supplement 1," Westinghouse Electric Company, April 2006. (ADAMS Accession No. ML061220450)

RAI 11-6

The NRC staff requested additional information regarding the application of POLCA-T to various cladding materials. The response states that the approach for POLCA-T is to incorporate specific material models and property correlations based on the **[**
]

In certain cases, the **[**

]

The NRC staff finds that this approach is acceptable to model the different cladding materials. The NRC staff has reviewed the validation of these models against test data in its review of the thermal-mechanical (T-M) design methodology. **[**

] Therefore, the scope of the POLCA-T application is limited to those fuel designs and cladding materials where the NRC staff has reviewed and approved the models and correlations.

The NRC staff requested additional information regarding the thermal expansion treatment of Zircaloy. The response states that the TR will be modified to remove statements regarding the anisotropy of thermal expansion. The response clarifies that the macro-behavior of the thermal expansion data from MATPRO shows that the **[**

] The NRC staff finds the response and proposed TR revision acceptable.

References

11-6.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

11-6.2 WCAP-15836-P-A, "Fuel Rod Design Methods for Boiling Water Reactors – Supplement 1," Westinghouse Electric Company, April 2006. (ADAMS Accession No. ML061220450)

RAI 11-7

The NRC staff requested additional information regarding the thermal expansion model for zirconium-based alloys. The response states that the model is based on the MATPRO materials properties developed mainly for Zircaloy-4, **[**

]

References

11-7.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

RAI 11-8

The NRC staff requested additional information regarding the cold work parameter. The response states that the cold work parameter is included to account for the effects of cold work on the Young and Shear moduli. The elastic moduli are primarily affected by temperature and oxygen content, and the cold work effects are much less important. Therefore, the response states that in cases where the cold work is not known, a default value of zero is used.

The response further provides comparisons of the predicted and measured moduli against zirconium, Zircaloy-2, and Zircaloy-4 data. The data include test data from Busby. The range of cold work considered in the Busby data varies between 0 percent and 25 percent. The data indicate that the impact of cold work is small and negligible compared to the model uncertainty. Therefore, the NRC staff finds that the use of the default value of 0 is acceptably accurate for reactor analyses. Similarly, the NRC staff finds that the model is based on relevant experimental data and including the known cold work will slightly improve accuracy. Therefore, the NRC staff finds that the response is acceptable.

Reference

11-8.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

RAI 11-9

The NRC staff requested additional information regarding the basis for the Poisson's ratio for isotropic materials. The NRC staff requested that Westinghouse consider the effects of forming processes on anisotropic properties such as texture. The response states that the Poisson's ratio calculation is based on the approved thermal-mechanical (T-M) methodology in WCAP-15836-P-A, which treats the Poisson's ratio as a function of the **[**

]

In the response to RAI 11-8, Westinghouse provided data from the MATPRO database confirming that the moduli are primarily a function of the temperature and oxygen content and that the cold work and texture are second order effects. **[**

]

Therefore, the NRC staff finds that the response is acceptable.

References

11-9.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747P, "POLCA-T: System Analysis Code with Three-Dimensional Core Model" (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

11-9.2 WCAP-15836-P-A, "Fuel Rod Design Methods for Boiling Water Reactors – Supplement 1," Westinghouse Electric Company, April 2006. (ADAMS Accession No. ML061220450)

RAI 11-10

The NRC staff requested additional information regarding the cladding creep model. The model is used in the prediction of the gas gap size, and hence in the calculation of the gap heat transfer and initial stored energy. The response states that Equation 14.70 is used to determine the contribution of tangential clad deformation to creep and elastic strains to gap size change.

The response states that the model has been verified against the approved T-M methodology described in WCAP-15836-P-A. The NRC staff finds that this is a reasonable basis for the model verification based on the NRC staff's approval of the methodology. Therefore, the NRC staff finds that the response is adequate to justify the applicability of the model.

Reference

11-10.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

RAI 11-11

The NRC staff requested that Westinghouse explain why cladding elastic deformation is modeled in only two dimensions. The response states that the cladding elastic deformation model is applied on a nodal basis. Therefore, the cladding elastic deformation is calculated in two dimensions for each node, and the axial variation is captured in axial nodalization of the core model. Therefore, the model calculates the elastic deformation in three dimensions. The NRC staff finds that this clarification is acceptable.

Reference

11-11.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-27, June 25, 2008. (ADAMS Accession No. ML081890191)

RAI 11-12

Accurate calculation of the metal-water reaction is not required for cases where significant core heat-up does not occur. Therefore, the NRC staff has deferred the review of the subject RAI response to the review of the application of POLCA-T to either ATWS simulations or LOCA. The NRC staff's review and approval of POLCA-T for CRDA and stability analyses does not constitute the NRC staff acceptance of the subject RAI response. This response contains information related to the response to RAI 9-1 regarding the heat addition from exothermic metal water reactions, and the basis for the NRC staff's deferral is identical.

The NRC staff notes, however, that the calculation of the initial oxide thickness falls within the scope of the subject review. The NRC staff notes that the accurate prediction of the initial cladding thickness is likely to have a direct effect on the efficacy of the stability methodology based on the sensitivity of the decay ratio to the fuel thermal time constant. The NRC staff has requested additional information in the area of the initial cladding thickness under RAI 4-11. The staff reviewed the response to RAI 4-11 and the detailed staff review is provided under a separate section in this Appendix.

Reference

11-12.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up Response to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-36, August 22, 2008. (ADAMS Accession No. ML082520770)

RAI 11-13

In RAI 11-13, the NRC staff requested additional qualification information and detailed explanation of the cladding rupture model. Qualification of the cladding rupture model against **[ ]** was provided in the response. The cladding rupture model is generally used in systems analysis codes to evaluate the core geometry changes under loss-of-coolant accident or ATWS scenarios in order to establish if the core retains a coolable geometry. For new reactor applications, such as application to the ABWR, an assessment of core coolability is required. The interim criteria in Appendix B of SRP 4.2 specifically develop core coolability criteria for CRDA that require determination of the cladding rupture.

Based on the information provided, the NRC staff could not reach a conclusion regarding the acceptability of the cladding rupture model in POLCA-T. Specifically, there appeared to be differences between the POLCA-T cladding rupture model and the previously-approved model. Secondly, the POLCA-T methodology does not appear to treat cladding rupture due to rod-to-rod contact. Therefore, the NRC staff issued a supplemental request for additional information.

The NRC staff requested additional information regarding the consistency of the cladding rupture model with the previously-approved cladding rupture model described in CENPD-293-P-A and WCAP-15682-P-A (References 11-13.3 and 11-13.4).

Supplemental Information Provided for Response to RAI 11-13S1

Westinghouse provided a response to RAI 11-13S1 in Reference 11-13.5. The NRC staff's supplemental RAI was divided into seven areas addressing the consistency of the cladding rupture model.

Part 1

The NRC staff requested that Westinghouse confirm that the cladding rupture model was fully consistent with the cladding rupture model approved by the NRC staff in WCAP-15682-P-A. The response confirms that the models are consistent. Therefore, the NRC staff finds that the use of the model is acceptable on the basis of its previous approval.

Part 2

The NRC staff noted several differences in the POLCA-T TR description of the cladding rupture model and the description provided in Reference 11-13.4. In response to RAI 11-13S1, Westinghouse provided corrections to typographical errors in the model description in the POLCA-T TR. These corrections bring the POLCA-T description into alignment with the model previously approved by the NRC staff. Therefore, the NRC staff finds that the corrections are appropriate.

Part 3

To ensure that the model is consistent, the NRC staff requested that Westinghouse provide a figure that is substantially similar to a qualification figure (Figure 7-22 in Reference 11-13.3). The figure in Reference 11-13.3 provides a comparison of the Westinghouse cladding rupture model against measured data in Reference 11-13.2. The response provided in Reference 11-13.5 contains similar figures as those in the previously approved TR CENPD-293-P-A.

The NRC staff reviewed the figures and found that **[**
]
These figures demonstrate that the cladding rupture model performance is consistent with the performance of the previously approved model. Therefore, the NRC staff is reasonably assured that the POLCA-T model is fully consistent with the previously approved model and, therefore, the response is acceptable.

Part 4

The NRC staff requested additional information regarding the phenomenon of rod-to-rod contact. The response states that POLCA-T accounts for the **[** **]** Therefore, the NRC staff finds that the response is acceptable as this phenomenon is accounted for.

Part 5

In its previous approval of the **[**
] The NRC staff requested that Westinghouse **[** **]** The response states that the **[**
] Therefore, the NRC staff finds that the response is acceptable.

Part 6

The NRC staff requested additional descriptive details regarding the double layer burst stress modifier. Specifically, the NRC staff requested that Westinghouse describe in greater detail how the double layer thickness and total oxygen concentration are determined. The response states that the use of Equation 14-85 is consistent with the TR description in Reference 11-13.3. The NRC staff finds that this approach is acceptable on the basis that it is fully consistent with the approved method.

The NRC staff reviewed the clarification provided in the RAI response and found that it was adequate for the NRC staff to confirm that the POLCA-T model is employed consistently with its previous approval in Reference 11-13.3. Therefore, its usage is acceptable.
Part 7

The NRC staff requested additional information regarding the uncertainty analysis. The response to the NRC staff's RAI states that the **[**

] Therefore, the NRC staff finds that the **[**
] The NRC staff defers the review of the cladding rupture model uncertainties to its review of the usage of this model for a specific application.

References

11-13.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Follow-Up Response to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-proprietary)," LTR-NRC-08-36, August 22, 2008. (ADAMS Accession No. ML082520770)

11-13.2 Powers, D., Meyer, R., "Cladding Swelling and Rupture Models for LOCA Analysis," NUREG-0630, March 1980.

11-13.3 CENPD-293-P-A, "BWR ECCS Evaluation Model: Supplement 1 to Code Description and Qualification," ABB CE, July 1996.

11-13.4 WCAP-15682-P-A, "Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application," Westinghouse Electric Company, May 2003. (ADAMS Accession No. ML031540688)

11-13.5 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Fourth Set of Responses to the Second Round of NRC's Request of Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-09-21, April 8, 2009. (ADAMS Accession No.: ML100281005)

RAI 11-14

The NRC staff requested additional information regarding the gas gap heat conduction model in POLCA-T. First, the NRC staff requested that Westinghouse describe the relationship between the Sutherland weighting factors and the weighting factors used in the STAV7.2 code. The response states that Chapter 20.3 of the original TR included misleading language and that the model presented in the TR is for calculating the thermal-hydraulic behavior of non-condensable gases in the coolant. The response provides a TR revision to include Chapter 20.4 which describes the models for properties of the gases in the gas gap.

The NRC staff reviewed the revised section provided in the RAI response. The revised TR describes a gas gap conduction model that is consistent with the approved STAV 7.2 models (Reference 11-14.1). The response also states that while this information was not presented in the TR, these models had been implemented since the original POLCA-T code release (Reference 11-14.1). The NRC staff reviewed the description of the gas gap conduction models and confirmed that they are consistent with the previously-approved models. On this basis, the NRC staff finds that the POLCA-T gas gap properties models are acceptable.

The response further corrects several editorial and typographical errors (Reference 11-14.1). The NRC staff reviewed the revisions and found that the corrections are consistent with the previously-approved models. Therefore, the NRC staff has confirmed that the POLCA-T models are consistent with those models that have previously been approved by the NRC staff. The response states that the corrections will be made to the approved TR. The NRC staff finds this acceptable.

Reference

11-14.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Fifth Set of Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-09-25, May 12, 2009. (ADAMS Accession No. ML091380095)

RAI 11-15

The NRC staff requested additional information regarding the translation of previously-approved fuel rod models to the POLCA-T code. In particular, the NRC staff requested additional information regarding the fuel thermal conductivity model and the gas gap pressure model. The

NRC staff additionally requested that Westinghouse provide a summary of those historically-approved models that have been directly translated into POLCA-T.

Fuel Thermal Conductivity

In response to the first portion of the NRC staff's RAI, Westinghouse revised the TR to incorporate fuel thermal conductivity and pellet relocation models that are consistent with WCAP-15836-P-A (STAV7.2) (Reference 11-15.1). The NRC staff reviewed the response and confirmed that the additional text is consistent with the approved models. The response further states that several STAV models are maintained in POLCA-T for consistency with other versions of STAV; this is to facilitate licensing evaluations across several nations. The response specifically states, however, that the equation set presented in WCAP-15836-P-A is from STAV7.2 and will be used in POLCA-T when performing licensing calculations in the U.S. The NRC staff finds that this is acceptable on the basis that these models are the most recently reviewed and approved models. The response further states that the if the STAV7.2 model is changed, then POLCA-T will also be updated with the new approved fuel rod models, and that the NRC's approval of any new T-M model is independent of POLCA-T and is licensed separately. The NRC staff agrees with the response. The response specifically references WCAP-15836-P-A as an example of the NRC's review and approval of the stand-alone fuel T-M methodology.

The NRC staff will impose the condition that U.S. licensing evaluations be performed using STAV7.2 or a subsequently approved T-M model. The NRC staff will likewise impose conditions on the update of POLCA-T to incorporate subsequently-approved models. These conditions are fully consistent with the RAI response.

Gas Gap Pressure

The NRC staff requested additional information regarding the incorporation of the plena in the calculation of the rod internal pressure. A key assumption in the **[**

]

The response includes Equation 14-30 which provides additional descriptive details of the calculation of the rod internal pressure. This calculation is based on an ideal gas treatment of the gap gases and accounts for the gas composition and the distribution of temperatures within the various sub-volumes comprising the rod internal void space (Reference 11-15.1).

The response also states that **[**

]

The NRC staff requested that Westinghouse **[**

] Overall, the NRC staff finds that this

approximation is reasonable for reactor systems analysis. **[**

]

While this approach **[**

] Therefore, the NRC staff finds that the approach for calculating the rod internal pressure is acceptable.

Translation of Previously Approved Models

The NRC staff noted in its review, several models in the TR in the description of the fuel rod model that appeared to be inconsistent with previously approved versions of the STAV or CHACHA codes. To assist the NRC staff in its review, Westinghouse provided a comprehensive listing of the individual models that comprise the overall fuel rod model. In each case, the response provides the reference to the specific equation in previously approved TRs that describes the corresponding model. The NRC staff has reviewed the associated historical TRs (CENPD-285-P-A, WCAP-15836-P-A, and CENPD-293-P-A), the subject TR and other RAI responses. Based on this review, the NRC staff has confirmed that the POLCA-T fuel rod model is consistent with the approved STAV models. The response states that the compilation of equations and their counterparts in other approved TRs demonstrates that all equations have approved references or are basic in nature or part of a derivation (Reference 11-15.1). The NRC staff agrees with this statement.

The response notes two deviations between the model description in the subject TR and the STAV7.2 model. These are the thermal conductivity model and the pellet relocation model. As is the subject of RAI 11-14, these models have been previously implemented in POLCA-T and will be used for U.S. licensing calculations. As stated in the response to RAI 11-14, the TR will be revised to include these STAV7.2 models (Reference 11-15.1). Therefore, the NRC staff concludes that the fuel rod model in POLCA-T is fully consistent with the most recently reviewed and approved fuel rod models developed by Westinghouse. Therefore, the NRC staff finds that the usage of these models within POLCA-T is acceptable.

Historical Limitations and Conditions

The NRC staff requested that Westinghouse consider the conditions and limitations that were imposed on STAV7.2 during the review of WCAP-15836-P-A. Specifically, the NRC staff requested that Westinghouse evaluate the conditions and limitations for dual applicability to POLCA-T, and in cases where the conditions were not applicable, that Westinghouse provide a detailed rationale.

The responses states that five conditions listed in WCAP-15836-P-A, as a result of the NRC review of the Westinghouse T-M modeling of BWR fuel performance, are acceptable for dynamic analyses, with one exception. This exception is the value of the Γ parameter. In all STAV versions prior to STAV7.2, the value of this parameter was **[**

] (Reference 11-15.1).

The Γ parameter is an azimuthally "misfitting" fraction between relocated pellet fragments and the clad. It accounts for increased heat transfer between pellets and the cladding where the cracked portions of the pellet relocate and come in contact with the cladding prior to gap closure during irradiation.

The response provides the results of several stability calculations performed using Γ values of **[ ]** The results indicate that DR and frequency are better predicted when the Γ value is **[ ]**

Based on the stability calculations, it was observed that setting Γ to **[ ]** is more appropriate for best-estimate transient and stability calculations. Therefore, the NRC staff finds Westinghouse has acceptably justified deviating from the specified value of Γ in WCAP-15836-P-A for dynamic evaluations.

References

11-15.1.1 Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, "Fifth Set of Responses to the Second Round of NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation for Topical Report (TR) WCAP-16747-P, 'POLCA-T: System Analysis Code with Three-Dimensional Core Model' (TAC No. MD5258) (Proprietary/Non-Proprietary)," LTR-NRC-09-25, May 12, 2009.

11-15.1.2 CENPD-293-P-A, "BWR ECCS Evaluation Model: Supplement 1 to Code Description and Qualification," ABB CE, July 1996.

11-15.1.3 WCAP-15682-P-A, "Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application," Westinghouse Electric Company, May 2003. (ADAMS Accession No. ML031540688)

11-15.2 CENPD-285-P-A, "Fuel Rod Design Methods for Boiling Water Reactors," ABB CE, July 1996.