Turkey Point Units 3 and 4
Extended Power Uprate
Licensing Report

Attachment 4
Appendix A

Safety Evaluation Report Compliance

This coversheet plus 24 pages

A.1 SAFETY EVALUATION REPORT COMPLIANCE INTRODUCTION

This Appendix is being provided as a supplement to the codes and methods information provided in LR Section 2.8.5.0 for the PTN Extended Power Uprate and is directly applicable to the PTN EPU analyses. This appendix addresses compliance with the limitations, restrictions, and conditions for the codes and methods listed below and specified in the approving safety evaluation of the applicable codes and methods (RS-001 Section 2.1 Matrix 8 Note 7).

Table A.1-1 presents an overview of the Safety Evaluation Reports (SER) by codes and methods. For each SER, the applicable report subsections and appendix subsections are listed.

Table A.1-1
Safety Evaluation Report Compliance Summary

No.	Subject	Topical Report (Reference)/ Date of NRC Acceptance	Code(s)	Limitation, Restriction, Condition	LR Section	Appendix Section
	Non-LOCA Thermal Transients	WCAP-7908-A (Reference A.1-1)/ September 30, 1986	FACTRAN	Yes	2.8.5.4.1 2.8.5.4.6	A.2
	Non-LOCA Safety Analysis	WCAP-14882-P-A (Reference A.1-2)/ February 11, 1999	RETRAN	Yes	2.8.5.1.1 2.8.5.1.2 2.8.5.2.1 2.8.5.2.2 2.8.5.2.3 2.8.5.3.1 2.8.5.3.2 2.8.5.4.2	A.3
	Non-LOCA Safety Analysis	WCAP-7907-P-A (Reference A.1-3)/ July 29, 1983	LOFTRAN	Yes	2.8.5.4.2 2.8.5.4.3 2.8.5.7	A.4
	Non-LOCA Thermal/ Hydraulics	WCAP-11397-P-A Reference A.1-14 January 17, 1989	RTDP	Yes	2.8.3 2.8.5.1.1 2.8.5.1.2 2.8.5.2.1 2.8.5.3.1 2.8.5.3.2 2.8.5.4.2 2.8.5.4.3	A.8
	Neutron Kinetics	WCAP-7979-P-A (Reference A.1-4)/ July 29, 1974	TWINKLE	None for Non-LOCA Transient Analysis	2.8.5.4.1 2.8.5.4.6	Not Applicable

Table A.1-1 (Continued)
Safety Evaluation Report Compliance Summary

No.	Subject	Topical Report (Reference)/ Date of NRC Acceptance	Code(s)	Limitation, Restriction, Condition	LR Section	Appendix Section
1.	Multi-dimensional Neutronics	WCAP-10965-P-A (Reference A.1-5)/ June 23, 1986	ANC	None for Non-LOCA Transient Analysis	2.8.5.1.2 2.8.5.4.3 2.8.2	Not Applicable
2.	Non-LOCA Thermal/ Hydraulics	WCAP-14565-P-A (Reference A.1-6)/ January 19, 1999	VIPRE	Yes	2.8.5.1.1 2.8.5.1.2 2.8.5.3.1 2.8.5.3.2 2.8.5.4.1 2.8.5.4.3 2.8.3	A.5
3.	Steam Generator Tube Rupture	WCAP-10698-P-A (Reference A.1-15)/ March 30, 1987	LOFTTR2	None for Steam Generator Tube Rupture	2.8.5.6.2	Not Applicable
4.	App K SBLOCA	WCAP-10079-P-A, WCAP-10054-P-A (with addenda), WCAP-11145, WCAP-14710 (References A.1-7 through A.1-11)/ May 23, 1985	NOTRUMP	Yes	2.8.5.6.3.3	A.6
5.	LOCA Hydraulic Forces	WCAP-8708-P-A (Reference A.1-12/ June 17, 1977, WCAP-9735 Rev. 2 (Reference A.1-13)	MULTIFLEX 3.0	Yes	2.8.5.6.3.5	A.7
6.	ASTRUM BELOCA	WCAP-16009-P-A (Reference A.1-16)/ November 5, 2004	WCOBRA/ TRAC	Yes	2.8.5.6.3.2	A.9

References

A.1-1 WCAP-7908-A, "FACTRAN – A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod," H. G. Hargrove, December 1989.

A.1-2 WCAP-14882-P-A, “RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses,” D. S. Huegel, et al., April 1999.

A.1-3 WCAP-7907-P-A, “LOFTRAN Code Description,” T. W. T. Burnett, et al., April 1984.

A.1-4 WCAP-7979-P-A, “TWINKLE – A Multi-Dimensional Neutron Kinetics Computer Code,” D. H. Risher, Jr. and R. F. Barry, January 1975.

A.1-5 WCAP-10965-P-A, “ANC: A Westinghouse Advanced Nodal Computer Code,” Y. S. Liu, et al., September 1986.

A.1-6 WCAP-14565-P-A, “VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,” Y. X. Sung, et al., October 1999.

A.1-7 WCAP-10079-P-A and WCAP-10080-A, “NOTRUMP - A Nodal Transient Small Break and General Network Code,” Meyer, P. E., August 1985.

A.1-8 WCAP-10054-P-A, “Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code,” N. Lee, et al., August 1985.

A.1-9 WCAP-10054-P-A, Addendum 2, Revision 1, “Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model,” C. M. Thompson, et al., July 1997.

A.1-10 WCAP-11145-P-A, “Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code,” S. D. Rupprecht, et al., 1986.

A.1-11 WCAP-14710-P-A, “1-D Heat Conduction Model for Annular Fuel Pellets,” D. J. Shimeck, May 1998.

A.1-12 WCAP-8708-P-A and WCAP-8709-A, “MULTIFLEX A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics,” K. Takeuchi, et al., September 1977.

A.1-13 WCAP-9735, Rev. 2 and WCAP-9736, Rev. 1, “MULTIFLEX 3.0 A FORTRAN IV Computer Program for Analyzing Thermal-Hydraulic-Structural System Dynamics Advanced Beam Model,” K. Takeuchi, et al., February 1998.

A.1-14 WCAP-11397-P-A, “Revised Thermal Design Procedure,” Friedland, A. J. and Ray, S., April 1999.

A.1-15 WCAP-10698-P-A and WCAP-10750-A, “SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill,” R. N. Lewis, et al., August 1987.

A.1-16 M. E. Nissley, et. al., Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM), WCAP-16009-P-A (Proprietary Version), WCAP-16009-NP-A (Non-Proprietary Version), January 2005.

A.2 FACTRAN FOR NON-LOCA THERMAL TRANSIENTS

Table A.2-1
FACTRAN for Non-LOCA Thermal Transients

Limitations, Restrictions, and Conditions
1. "The fuel volume-averaged temperature or surface temperature can be chosen at a desired value which includes conservatisms reviewed and approved by the NRC."
Justification
The FACTRAN code was used in the analyses of the following transients for PTN: Uncontrolled Rod Withdrawal from Subcritical (PTN UFSAR Section 14.1.1) and RCCA Ejection (PTN UFSAR Section 14.2.6). Initial fuel temperatures used as FACTRAN input in the RCCA Ejection analysis were calculated using the NRC-approved PAD 4.0 computer code, as described in WCAP-15063-P-A (Reference A.2-1). As indicated in WCAP-15063-P-A, the NRC has approved the method of determining uncertainties for PAD 4.0 fuel temperatures.
2. "Table 2 presents the guidelines used to select initial temperatures."
Justification
In summary, Table 2 of the SER specifies that the initial fuel temperatures assumed in the FACTRAN analyses of the following transients should be "High" and include uncertainties: loss of flow, locked rotor, and rod ejection. As discussed above, fuel temperatures were used as input to the FACTRAN code in the RCCA ejection analysis for PTN. The assumed fuel temperatures, which were calculated using the PAD 4.0 computer code (Reference A.2-1), include uncertainties and are conservatively high. FACTRAN was not used in the loss of flow and locked rotor analyses.
3. "The gap heat transfer coefficient may be held at the initial constant value or can be varied as a function of time as specified in the input."
Justification
The gap heat transfer coefficients applied in the FACTRAN analyses are consistent with SER Table 2. For the rod withdrawal from subcritical transient, the gap heat transfer coefficient is kept at a conservative constant value throughout the transient; a high constant value is assumed to maximize the peak heat flux (for DNB concerns) and a low constant value is assumed to maximize fuel temperatures. For the RCCA ejection transient, the initial gap heat transfer coefficient is based on the predicted initial fuel surface temperature, and is ramped rapidly to a very high value at the beginning of the transient to simulate clad collapse onto the fuel pellet.

Table A.2-1 (Continued)
FACTRAN for Non-LOCA Thermal Transients

Limitations, Restrictions, and Conditions
4. "…the Bishop-Sandberg-Tong correlation is sufficiently conservative and can be used in the FACTRAN code. It should be cautioned that since these correlations are applicable for local conditions only, it is necessary to use input to the FACTRAN code which reflects the local conditions. If the input values reflecting average conditions are used, there must be sufficient conservatism in the input values to make the overall method conservative."
Justification Local conditions related to temperature, heat flux, peaking factors and channel information were input to FACTRAN for each transient analyzed for PTN {Uncontrolled rod withdrawal from subcritical (PTN UFSAR Section 14.1.1) and RCCA ejection (PTN UFSAR Section 14.2.6)}. Therefore, additional justification is not required.
5. "The fuel rod is divided into a number of concentric rings. The maximum number of rings used to represent the fuel is 10. Based on our audit calculations we require that the minimum of 6 should be used in the analyses."
Justification At least 6 concentric rings were assumed in FACTRAN for each transient analyzed for PTN (Uncontrolled rod withdrawal from subcritical (PTN UFSAR Section 14.1.1) and RCCA ejection (PTN UFSAR Section 14.2.6.
6. "Although time-independent mechanical behavior (e.g., thermal expansion, elastic deformation) of the cladding are considered in FACTRAN, time-dependent mechanical behavior (e.g., plastic deformation) is not considered in the code. …for those events in which the FACTRAN code is applied (see Table 1), significant time-dependent deformation of the cladding is not expected to occur due to the short duration of these events or low cladding temperatures involved (where DNBR Limits apply), or the gap heat transfer coefficient is adjusted to a high value to simulate clad collapse onto the fuel pellet."
Justification The two transients that were analyzed with FACTRAN for PTN (Uncontrolled rod withdrawal from subcritical (PTN UFSAR Section 14.1.1) and RCCA ejection (PTN UFSAR Section 14.2.6)) are included in the list of transients provided in Table 1 of the SER; each of these transients is of short duration. For the Uncontrolled rod withdrawal from subcritical transient, relatively low cladding temperatures are involved, and the gap heat transfer coefficient is kept constant throughout the transient. For the RCCA ejection transient, a high gap heat transfer coefficient is applied to simulate clad collapse onto the fuel pellet. The gap heat transfer coefficients applied in the FACTRAN analyses are consistent with SER Table 2.

Table A.2-1 (Continued)
FACTRAN for Non-LOCA Thermal Transients

Limitations, Restrictions, and Conditions
7. "The one group diffusion theory model in the FACTRAN code slightly overestimates at beginning of life (BOL) and underestimates at end of life (EOL) the magnitude of flux depression in the fuel when compared to the LASER code predictions for the same fuel enrichment. The LASER code uses transport theory. There is a difference of about 3 percent in the flux depression calculated using these two codes. When [T(centerline) – T(Surface)] is on the order of 3000°F, which can occur at the hot spot, the difference between the two codes will give an error of 100°F. When the fuel surface temperature is fixed, this will result in a 100°F lower prediction of the centerline temperature in FACTRAN. We have indicated this apparent nonconservatism to Westinghouse. In the letter NS-TMA-2026, dated January 12, 1979, Westinghouse proposed to incorporate the LASER-calculated power distribution shapes in FACTRAN to eliminate this non-conservatism. We find the use of the LASER-calculated power distribution in the FACTRAN code acceptable."
Justification The condition of concern (T(centerline) – T(surface) on the order of 3000°F) is expected for transients that reach, or come close to, the fuel melt temperature. As this applies only to the RCCA ejection transient, the LASER-calculated power distributions were used in the FACTRAN analysis of the RCCA ejection transient for PTN.
List of transients and accidents that use the FACTRAN program (approved in NRC SER) A. Uncontrolled RCC Assembly Bank Withdrawal from a Subcritical Condition. B. Partial Loss of Forced Reactor Coolant Flow C. Complete Loss of Forced Reactor Coolant Flow. D. Single Reactor Coolant Pump Locked Rotor. E. Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection) For the PTN EPU, FACTRAN was used for A and E from the approved list.

References

A.2-1 WCAP-15063-P-A, Revision 1 (with Errata) "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," J. P. Foster and S. Sidener, July 2000.

A.3 RETRAN FOR NON-LOCA SAFETY ANALYSIS

Table A.3-1
RETRAN for Non-LOCA Safety Analysis

Limitations, Restrictions, and Conditions
1. "The transients and accidents that Westinghouse proposes to analyze with RETRAN are listed in this SER (Table 1) and the NRC staff review of RETRAN usage by Westinghouse was limited to this set. Use of the code for other analytical purposes will require additional justification."
Justification The transients listed in Table 1 of the SER are: • Feedwater system malfunctions • Excessive increase in steam flow • Inadvertent opening of a steam generator relief or safety valve • Steam line break • Loss of external load/turbine trip • Loss of offsite power • Loss of normal feedwater flow • Feedwater line rupture • Loss of forced reactor coolant flow • Locked reactor coolant pump rotor/sheared shaft • Control rod cluster withdrawal at power • Dropped control rod cluster/dropped control bank • Inadvertent increase in coolant inventory • Inadvertent opening of a pressurizer relief or safety valve • Steam generator tube rupture The transients analyzed for PTN using RETRAN are: • Feedwater system malfunctions • Excessive increase in steam flow • Steam line break • Loss of external electrical load/Turbine trip • Loss of all alternating current power to the station auxiliaries • Loss of normal feedwater flow • Loss of reactor coolant flow • Locked rotor accident • Uncontrolled rod withdrawal at power As each transient analyzed for PTN using RETRAN matches one of the transients listed in Table 1 of the SER, additional justification is not required.

Table A.3-1 (Continued)
RETRAN for Non-LOCA Safety Analysis

Limitations, Restrictions, and Conditions
2. "WCAP-14882 describes modeling of Westinghouse designed 4-, 3, and 2-loop plants of the type that are currently operating. Use of the code to analyze other designs, including the Westinghouse AP600, will require additional justification."
Justification
The PTN consists of two 3-loop Westinghouse-designed units that were "currently operating" at the time the SER was written (February 11, 1999). Therefore, additional justification is not required.
3. "Conservative safety analyses using RETRAN are dependent on the selection of conservative input. Acceptable methodology for developing plant-specific input is discussed in WCAP-14882 and in Reference 14 [WCAP-9272-P-A]. Licensing applications using RETRAN should include the source of and justification for the input data used in the analysis."
Justification
The input data used in the RETRAN analyses performed by Westinghouse came from both PTN and Westinghouse sources. Assurance that the RETRAN input data is conservative for PTN is provided via Westinghouse's use of transient-specific analysis guidance documents. Each analysis guidance document provides a description of the subject transient, a discussion of the plant protection systems that are expected to function, a list of the applicable event acceptance criteria, a list of the analysis input assumptions (e.g., directions of conservatism for initial condition values), a detailed description of the transient model development method, and a discussion of the expected transient analysis results. Based on the analysis guidance documents, conservative plant-specific input values were requested and collected from the responsible PTN and Westinghouse sources. Consistent with the Westinghouse Reload Evaluation Methodology described in WCAP-9272-P-A (Reference A.3-1), the safety analysis input values used in the PTN analyses were selected to conservatively bound the values expected in subsequent operating cycles.

References

A.3-1 WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," S. L. Davidson (Ed.), July 1985.

A.4 LOFTRAN FOR NON-LOCA SAFETY ANALYSIS

Table A.4-1
LOFTRAN for Non-LOCA Safety Analysis

Limitations, Restrictions, and Conditions
1. "LOFTRAN is used to simulate plant response to many of the postulated events reported in Chapter 15 of PSARs and FSARs, to simulate anticipated transients without scram, for equipment sizing studies, and to define mass/energy releases for containment pressure analysis. The Chapter 15 events analyzed with LOFTRAN are: ***• Feedwater System Malfunction*** ***• Excessive Increase in Steam Flow*** ***• Inadvertent Opening of a Steam Generator Relief or Safety Valve*** ***• Steamline Break*** ***• Loss of External Load*** ***• Loss of Offsite Power*** ***• Loss of Normal Feedwater*** ***• Feedwater Line Rupture*** ***• Loss of Forced Reactor Coolant Flow*** ***• Locked Pump Rotor*** ***• Rod Withdrawal at Power*** ***• Rod Drop*** ***• Startup of an Inactive Pump*** ***• Inadvertent ECCS Actuation*** ***• Inadvertent Opening of a Pressurizer Relief or Safety Valve*** ***This review is limited to the use of LOFTRAN for the licensee safety analyses of the Chapter 15 events listed above, and for a steam generator tube rupture…"***
Compliance For PTN, the LOFTRAN code was used in the analyses of the rod cluster control assembly drop transient, ATWS (loss of external load and loss of normal feedwater), and the uncontrolled rod withdrawal at power primary overpressurization case. As each of these transients matches one of the transients listed in the SER, additional justification is not required.

A.5 VIPRE FOR NON-LOCA THERMAL/HYDRAULICS

Table A.5-1
VIPRE for Non-LOCA Thermal/Hydraulics

Limitations, Restrictions and Conditions
1. "Selection of the appropriate CHF correlation, DNBR limit, engineered hot channel factors for enthalpy rise and other fuel-dependent parameters for a specific plant application should be justified with each submittal."
Compliance The WRB-1 correlation with a 95/95 correlation limit of 1.17 was used in the DNB analyses for the Turkey Point 15x15 DRFA and 15x15 Upgrade fuel. The use of the WRB-1 DNB correlation was approved in WCAP-8762-P-A (Reference A.5-2). Applicability of WRB-1 to Upgrade fuel was established through the Fuel Criterion Evaluation Process (FCEP) in LTR-NRC-04-8 (Reference A.5-3). For conditions where WRB-1 is not applicable, analyses were performed using approved secondary CHF correlations (such as ABB-NV and WLOP) in compliance with the SER conditions licensed for use in the VIPRE code. (WCAP-14565-P-A and its Addendum 2-P-A, Reference A.5-4). The use of the plant specific hot channel factors and other fuel dependent parameters in the DNB analysis for the Turkey Point 15x15 fuel were justified using the same methodologies as for previously approved safety evaluations of other Westinghouse three-loop plants using the same fuel design.
2. "Reactor core boundary conditions determined using other computer codes are generally input into VIPRE for reactor transient analyses. These inputs include core inlet coolant flow and enthalpy, core average power, power shape and nuclear peaking factors. These inputs should be justified as conservative for each use of VIPRE."
Justification The core boundary conditions for the VIPRE calculations for the 15x15 fuel are all generated from NRC-approved codes and analysis methodologies. Conservative reactor core boundary conditions were justified for use as input to VIPRE. Continued applicability of the input assumptions is verified on a cycle-by-cycle basis using the Westinghouse reload methodology described in WCAP-9272-P-A (Reference A.5-1).
3. "The NRC Staff's generic SER for VIPRE set requirements for use of new CHF correlations with VIPRE. Westinghouse has met these requirements for using WRB-1, WRB-2 and WRB-2M correlations. The DNBR limit for WRB-1 and WRB-2 is 1.17. The WRB-2M correlation has a DNBR limit of 1.14. Use of other CHF correlations not currently included in VIPRE will require additional justification."
Justification As discussed in response to Condition 1, the WRB-1 correlation with a limit of 1.17 was used as the primary correlation in the DNB analyses of 15x15 DRFA and Upgrade fuel for Turkey Point. For conditions where WRB-1 is not applicable, analyses were performed using approved secondary CHF correlations licensed for the VIPRE code in Reference A.5-4.

Table A.5-1 (Continued)
VIPRE for Non-LOCA Thermal/Hydraulics

Limitations, Restrictions and Conditions
4. "Westinghouse proposes to use the VIPRE code to evaluate fuel performance following postulated design-basis accidents, including beyond-CHF heat transfer conditions. These evaluations are necessary to evaluate the extent of core damage and to ensure that the core maintains a coolable geometry in the evaluation of certain accident scenarios. The NRC Staff's generic review of VIPRE did not extend to post CHF calculations. VIPRE does not model the time-dependent physical changes that may occur within the fuel rods at elevated temperatures. Westinghouse proposes to use conservative input in order to account for these effects. The NRC Staff requires that appropriate justification be submitted with each usage of VIPRE in the post-CHF region to ensure that conservative results are obtained."
Justification
For application to Turkey Point safety analysis, the use of VIPRE in the post-critical heat flux region is limited to the peak clad temperature calculation for the locked rotor transient. The calculation demonstrated that the peak clad temperature in the reactor core is well below the allowable limit to prevent clad embrittlement. VIPRE modeling of the fuel rod is consistent with the model described in WCAP-14565-P-A and included the following conservative assumptions: • DNB was assumed to occur at the beginning of the transient, • Film boiling was calculated using the Bishop-Sandberg-Tong correlation, • The Baker-Just correlation accounted for heat generation in fuel cladding due to zirconium-water reaction. Conservative results were further ensured with the following input: • Fuel rod input based on the maximum fuel temperature at the given power, • The hot spot power factor was equal to or greater than the design linear heat rate, • Uncertainties were applied to the initial operating conditions in the limiting direction.

References

A.5-1 WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," S. L. Davidson (Ed.), July 1985.

A.5-2 WCAP-8762-P-A, "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," F. E. Motley et. al., July 1984.

A.5-3 LTR-NRC-04-8, "Fuel Criterion Evaluation Process (FCEP) Notification of the 15x15 Upgrade Design (Proprietary/Non-Proprietary)," James A. Gresham, February 6, 2004.

A.5-4 WCAP-14565-P-A Addendum 2-P-A, "Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications," A. Leidich, et. al., April 2008.

A.6 NOTRUMP FOR SMALL BREAK LOCA

NOTRUMP SER Restriction Compliance Summary

The following table contains a synopsis of the NRC imposed Safety Evaluation Report (SER) restrictions/requirements and the Westinghouse compliance status related to these issues. Not all the items identified are clearly SER restrictions, but sometimes state the NRC's interpretation of the Westinghouse Evaluation Methodology utilized for a particular aspect of the Small Break Loss Of Coolant Accident (LOCA) Evaluation Model.

Table A.6-1
WCAP-10054-P-A and WCAP-10079-P-A (References A.6-1 and A.6-2)

Limitations, Restrictions, and Conditions
WCAP-10054-P-A is titled "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," and is dated August 1985. The following summarizes the SER restrictions and requirements associated with this WCAP:
1. SER Wording (Page 6)
"The use of a single momentum equation implies that the inertias of the separate phases can not be treated. The model therefore would not be appropriate for situations when separate inertial effects are significant. For the small break transients, these effects are not significant."
SER Compliance
Inherent compliance due to the use of a single momentum equation.
2. SER Wording (Page 8)
"To assure the validity of this application, the bubble diameter should be on the order of 10^{-1}-2 cm. As long as steam generator tube uncovery (concurrent with a severe depressurization rate) does not occur, this option is acceptable."
SER Compliance
Westinghouse complies with this restriction for all Appendix K licensing basis calculations. Typical Appendix K calculations do not undergo a significant secondary side system depressurization in conjunction with steam generator tube uncovery due to the modeling methodology utilized.
3. SER Wording (Page 14)
"The two phase multiplier used is the Thom modification of the Martinelli-Nelson correlation. This model is acceptable per 10CFR50 Appendix K for LOCA analysis at pressure above 250 psia."
SER Compliance
The original NOTRUMP model was limited to no less than 250 psia since the model, as contained in the NOTRUMP code, did not contain information below this range. Westinghouse extended the model to below 250 psia, as allowed by Appendix K paragraph I-C-2, and reported these modifications to the NRC via the 1995 annual reporting period (NSD-NRC-96-4639).

Table A.6-1 (Continued)
WCAP-10054-P-A and WCAP-10079-P-A (References A.6-1 and A.6-2)

Limitations, Restrictions, and Conditions
4. SER Wording (Page 16)
"Westinghouse, however, has stated that the separator models are not used in their SBLOCA analyses."
SER Compliance
Westinghouse does not model the separators in the secondary side of the steam generators for Appendix K Small Break LOCA analyses; therefore, compliance exists.
5. SER Wording (Pages 16-17)
"Axial heat conduction is not modeled." and "Deletion of clad axial heat conduction maximizes the peak clad temperature."
SER Compliance
The Westinghouse Small Break LOCA is comprised of two computer codes, the NOTRUMP code which performs the detailed system wide thermal hydraulic calculations and the LOCTA code which performs the detailed fuel rod heatup calculations. The NOTRUMP code does not model axial conduction in the fuel rod and therefore complies. The LOCTA code has always accounted for axial conduction as is clearly stated in WCAP-14710-P-A which supplements the original NOTRUMP documentation.
6. SER Wording (Page 17)
"...; critical heat flux, W-2, W-3, or Macbeth, or GE transient CHF (the W-2 and W-3 correlations are used for licensing evaluations);..."
SER Compliance
The information presented here indicates that the NRC apparently misstated that Westinghouse was utilizing the W-2,W-3 correlations for Critical Heat Flux (CHF) in the fuel rod heat transfer model. A review of the analyses performed by Westinghouse, including those in WCAP-11145-P-A, indicates that the Macbeth CHF correlation has been utilized for all Appendix K analyses performed by Westinghouse. This is consistent with the slab heat transfer map as described in WCAP-10054-P-A. In addition, the Macbeth correlation is specifically called out in Appendix K I-C-4-4 as an acceptable CHF model. In a supplemental response to NRC questions (Specifically question 440.1 found in Appendix A of WCAP-10054-P-A, Page A-10), a description of the core model describes the Macbeth as being utilized as the CHF correlation in the NOTRUMP Small Break LOCA model.
7. SER Wording (Page 21)
"The standard continuous contact model is not appropriate for vertical flow,..."
SER Compliance
The standard continuous contact flow links are not utilized when modeling vertical flow in the Appendix K NOTRUMP Evaluation Model analyses; therefore, compliance is demonstrated.

Table A.6-1 (Continued)
WCAP-10054-P-A and WCAP-10079-P-A (References A.6-1 and A.6-2)

Limitations, Restrictions, and Conditions
8. SER Wording (Page 27)
"..., the hardwired choice of one fuel pin time step per coolant time step should result in sufficient accuracy."
SER Compliance
The NOTRUMP code continues to utilize only one fuel pin time step per coolant time step and therefore complies with this requirement.
9. SER Wording (Page 47)
"The code options available to the user but not applied in licensing evaluations were not reviewed."
SER Compliance
Westinghouse complies with this requirement.
10. SER Wording (Page 53)
"4. Steam Interaction with ECCS Water, a. Zero Steam Flow in the Intact Loops While Accumulators Discharge Water."
SER Compliance
Per paragraph I-D-4 Appendix K, the following is stated: "During refill and reflood, the calculated steam flow in unbroken reactor coolant pipes shall be taken to be zero during the time that accumulators are discharging water into those pipes unless experimental evidence is available regarding the realistic thermal-hydraulic interaction between the steam and the liquid. In this case, the experimental data may be used to support an alternate assumption." As can be seen, the specific Appendix K wording can be considered applicable to Large Break LOCAs only since Small Break LOCAs do not undergo a true refill/reflood period. However, the Westinghouse Small Break LOCA Evaluation Model methodology is such that for break sizes in which the intact loop seal restriction is not removed (WCAP-11145-P-A Page 2-11), steam flow through the intact loop(s) is automatically (artificially) restricted via the loop seal model. While not specifically limited to zero, the flow is drastically reduced via the application of the artificial loop seal restriction model. For breaks sizes above which the loop seal restriction is removed (typically ≥ 6 inch diameter breaks), this criterion is not explicitly adhered to. The implementation of the COSI condensation model into NOTRUMP (As approved by the NRC in WCAP-10054-P-A, Addendum 2, Revision 1), which is based on additional experimental documentation and improved modeling techniques, more accurately models the interaction of steam with Emergency Core Cooling Water in the cold leg region. This experimental documentation supports the more accurate modeling of steam/water interaction in the cold leg region as allowed by Appendix K. Note however that even with the COSI condensation model active, the accumulator injection condensation model still utilizes the conservative model as originally licensed in the NOTRUMP code.

Table A.6-1 (Continued)
WCAP-10054-P-A and WCAP-10079-P-A (References A.6-1 and A.6-2)

Limitations, Restrictions, and Conditions
11. SER Wording (Page 7 of enclosure 2)
"Per generic letter 83-35, compliance with Action Item II.K.3.31 may be submitted generically. We require that the generic submittal include validation that the limiting break location has not shifted away from the cold legs to the hot or pump suction legs."
SER Compliance
Westinghouse submitted WCAP-11145-P-A in support of generic letter 83-35 Action Item II.K.3.31. As part of this effort, verification was provided which documented that the cold leg break location remains limiting.
WCAP-10054-P-A, Addendum 2, Revision 1 (Reference A.6-3)
WCAP- 10054-P-A, Addendum 2, Revision 1 is titled "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," and is dated July 1997. The following summarizes the SER restrictions and requirements associated with this WCAP:
1. SER Wording (Page 3)
"It is stated in Ref. 5 that the range of injection jet velocities used in the experiments brackets the corresponding rates in small break LOCAs for Westinghouse plants and that the model will be used within the experimental range. Also in References 1 and 5 Westinghouse submitted analyses demonstrating that the condensation efficiency is virtually independent of RCS pressure and state that the COSI model will be applied within the pressure range of 550 to 1200 psia."
SER Compliance
The coding implementation of the COSI model correlation in the NOTRUMP model restricts the application of the COSI condensation model to a default pressure range of 550 to 1200 psia and limits the injection flow rate to a default value of 40 lbm/sec-loop. The value of 40 lbm/sec-loop corresponds to the 30 ft./sec velocity utilized in the COSI experiments. As such, the default NOTRUMP implementation of the COSI condensation model complies with the applicable SER restrictions.
WCAP-11145-P-A (Reference A.6-4)
WCAP-11145-P-A, is titled "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study With The NOTRUMP Code," and is dated 1986. No specific SER restrictions were provided by the NRC as part of this WCAP review; however, the SER contains verification that the requirements of Item II.K.3.31 have been satisfied (i.e. break location study).

Table A.6-1 (Continued)
WCAP-10054-P-A and WCAP-10079-P-A (References A.6-1 and A.6-2)

Limitations, Restrictions, and Conditions
1. SER Wording (Page 5)
"We therefore, find that the requirements of NUREG-0737, Item I.K3.31, as clarified by Generic Letter 83-35, have been satisfied."
SER Compliance
"We find that a condition of the safety evaluation for NOTRUMP as applied to Item II.K.3.30 has been satisfied. The limiting cold leg break size for a 4-loop plant was reanalyzed at pump suction and at hot leg locations. The results confirmed that the cold leg break was limiting."
WCAP-14710-P-A (Reference A.6-4)
WCAP-14710-P-A, is titled "1-D Heat Conduction Model for Annular Fuel Pellets," and is dated May 1998. No specific SER restrictions are provided by the NRC in this document; however, a conclusion was reached regarding the modeling of annular pellets during Small Break LOCA event.
1. SER Wording
"Based on its conclusions that the explicit modeling of annular pellets, as described in WCAP -4710(P), provides a more realistic representation in W Appendix K ECCS evaluation models of the annular pellets, while retaining conservatism in those evaluation models, the staff finds that the explicit modeling of annular pellets, as described in WCAP-14710(P), in W Appendix K LOCA evaluation models permits those models to continue to satisfy the regulations to which they were approved, and is, therefore, acceptable for incorporation into those models."
SER Compliance
Westinghouse performs sensitivity studies to assess the impact of modeling annular pellets on plant specific analyses.

References

A.6-1 WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," N. Lee, et al., August 1985.

A.6-2 WCAP-10079-P-A and WCAP-10080-A, "NOTRUMP - A Nodal Transient Small Break And General Network Code," Meyer, P. E., August 1985.

A.6-3 WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," C. M. Thompson, et al., July 1997.

A.6-4 WCAP-11145-P-A, "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code," S. D. Rupprecht, et al., 1986.

A.6-5 WCAP-14710-P-A, "1-D Heat Conduction Model for Annular Fuel Pellets," D. J. Shimeck, May 1998.

A.7 MULTIFLEX FOR LOCA HYDRAULIC FORCES

The NRC Safety Evaluation Report (SER) for the MULTIFLEX 1.0 Evaluation Model can be found in the front of WCAP-8708 Rev. 2 (Reference A.7-1). This SER stipulates a number of conditions and limitations on the use of the MULTIFLEX 1.0 Evaluation Model for licensing basis calculations. The following is a review of these SER restrictions and requirements.

Table A.7-2
MULTIFLEX 1.0

Limitations, Restrictions and Conditions	
1.	**SER Restriction** - Use of Corrected Sonic Velocity (SER, page 11) ***SER Wording - "The sonic velocity, or wave speed, computed with the empirical equation of state was not consistent with the 1967 ASME Steam Tables. The corrected sonic velocity data is required for a licensing calculation."***
	SER Compliance - The MULTIFLEX code has been changed (prior to the issuance of Revision 1 to WCAP-8708) to compute revised sonic velocity. Therefore, Westinghouse is in compliance with this restriction.
2.	**SER Restriction** - Lower Plenum Modeling (SER, page 12) ***SER Wording - "In the modeling region from the downcomer annulus to the lower plenum, the equivalent pipe network provided an artificially short transport distance across the length of the lower plenum. The correct radial transport distance, the diameter of the pressure vessel, is required in the model for a licensing calculation."***
	SER Compliance - Westinghouse does not use the "artificially short" lower plenum length cited in the SER. Therefore, it can be concluded that Westinghouse is in compliance with this modeling requirement.
3.	**SER Restriction** - 10 Mass Point Downcomer (SER, page 12, 18, 19) ***SER Wording - "The peak lateral force for a calculation using a 10 mass point representation for the core support barrel shows an increase in loading of 4% over the reference 5 mass point case. The NRC, therefore, requires a 10 mass point model be used for a coupled licensing calculation."***
	SER Compliance - Standard methodology uses a 10 mass point structural model. Therefore, Westinghouse is in compliance with this requirement.
4.	**SER Restriction** - 1 Millisecond Break Opening Time (BOT) (SER, page 13) ***SER Wording - "The use of a one millisecond opening time, as specified by Westinghouse, is required for a licensing calculation. Longer break opening times will not be considered unless Westinghouse demonstrated that the proposed break opening time with current equivalent pipe network adequately predicts the results of applicable experimental data."***
	SER Compliance - Standard methodology uses a 1 millisecond BOT. Therefore, Westinghouse is in compliance with this restriction.

Table A.7-2 (Continued)
MULTIFLEX 1.0

Limitations, Restrictions and Conditions	
5.	**SER Restriction** - Use of "Question 18" Input Parameters (SER, page 12). Question 18 establishes a line-by-line review of MULTIFLEX input. Parameters, identifying those that are "Required for design basis blowdown analysis" ***SER Wording - "The response to Question 18 of reference 4 is to be included in the MULTIFLEX report to identify the acceptable input option for a licensing calculation."***
	SER Compliance - The inputs used in the response to Question 18 were reviewed against the MULTIFLEX inputs established as Westinghouse's current methodology. We can state that our current models conservatively bound the requirements for licensing basis calculations as described in the MULTIFLEX SER. Therefore, Westinghouse is in compliance with this restriction.

MULTIFLEX 3.0 Applications

As indicated in the SER of WCAP-15029-P-A (Reference A.7-3), the WCAP-9735, Rev. 2 (Reference A.7-2) topical was submitted for NRC review and subsequently withdrawn. As stated in the SER, "Evaluation of the MULTIFLEX 3.0 methodology is not a requisite for concluding that WCAP-15029 is acceptable." The Staff's discussion of MULTIFLEX 3.0 is shown below:

"The MULTIFLEX 3.0 program is described as a more sophisticated analysis tool for LOCA hydraulic force calculations than the currently approved version, MULTIFLEX 1.0. WCAP-15029 indicates that the MULTIFLEX 3.0 program enhancements of MULTIFLEX 1.0 include: the use of a two dimensional flow network to represent the vessel downcomer region in lieu of a collection of one dimensional parallel pipes; the allowance for non-linear boundary conditions at the vessel and downcomer interface at the radial keys and the upper core barrel flange in lieu of simplified linear boundary conditions; and the allowance for vessel motion in lieu of rigid vessel assumptions. WCAP-15029 indicates that these modifications are included in the MULTIFLEX 3.0 program that is used to estimate the LOCA hydraulic forces on the vessel and consequential forces induced on the fuel and reactor vessel internal structures. The staff concurs with the WOG that MULTIFLEX 3.0 provides a more accurate and realistic modeling approach. On this basis, and considering that MULTIFLEX 3.0 is based on the previously approved MULTIFLEX 1.0, the staff considers the application of MULTIFLEX 3.0 with the WCAP-15029 methodology reasonable and acceptable."

Only one of the four SER restrictions in WCAP-15029-P-A (Reference A.7-3) applies to analyses performed using MULTIFLEX 3.0. Limitation number 2 reads: "The noding to be used in the representation of the loading is demonstrated to be adequate by performing nodalization sensitivity studies or by some other acceptable methodology."

The current nodalization employed in the Westinghouse baffle-former bolting analyses has been validated through a series of calculations. Westinghouse has verified that the current MULTIFLEX code version produces equivalent results to those used in the original development of MULTIFLEX 3.0 modeling features, despite several changes in operating system and computer platform. Westinghouse has demonstrated that the current standard nodalization produces equivalent results to those used in original test cases. Westinghouse has performed a series of sensitivity studies on MULTIFLEX 3.0 models using the current nodalization. Also, the historical model validation cases were found to yield conservative results relative to test data. This collection of documentation supports the conclusion that analyses performed to the current nodalization meet the limitation in WCAP-15029-P-A (Reference A.7-3).

MULTIFLEX 3.0 has also been accepted for use in other applications which are limited by the same acceptance criteria, i.e. fuel qualification. The Control Rod Insertion program, documented in WCAP-15245 (Reference A.7-4), was performed using MULTIFLEX 3.0 and the analyses were reviewed and accepted by the Staff (Reference A.7-5). These analyses have been used as a template for additional applications limited by the same acceptance criteria.

The use of break opening times greater than 1 millisecond has also been approved by the US-NRC (Reference A.7-6) for baffle barrel-bolting analyses. However, the use of longer break opening times is not approved for use on a generic basis. Such applications will require additional justification.

References

A.7-1 WCAP-8708-P-A and WCAP-8709-A, “MULTIFLEX A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics,” K. Takeuchi, et al., September 1977.

A.7-2 WCAP-9735, Rev. 2 and WCAP-9736, Rev. 1, “MULTIFLEX 3.0 A FORTRAN IV Computer Program for Analyzing Thermal-Hydraulic-Structural System Dynamics Advanced Beam Model,” K. Takeuchi, et al., February 1998.

A.7-3 WCAP-15029-P-A, WCAP-15030-NP-A, “Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions,” December 1998.

A.7-4 WCAP-15245 (Proprietary), WCAP-15246 (Non-proprietary), “Control Rod Insertion Following a Cold Leg LBLOCA, D. C. Cook, Units 1 and 2,” May 28, 1999.

A.7-5 Letter from John F. Stang (US-NRC) to Robert P. Powers (Indiana Michigan Power Company), “Issuance of Amendments - Donald C. Cook Nuclear Plant, Units 1 and 2 (TAC Nos. MA6473 and MA6474),” December 23, 1999.

A.7-6 WCAP-14748-P-A, Revision 0, WCAP-14749-NP-A, Revision 0, “Justification for Increasing Postulated Break Opening Times in Westinghouse Pressurized Water Reactors,” December 1998.

A.8 REVISED THERMAL DESIGN PROCEDURE FOR NON-LOCA THERMAL HYDRAULICS

Table A.8-1
Revised Thermal Design Procedure for Non-LOCA Thermal Hydraulics

Limitations, Restrictions, and Conditions
1. "Sensitivity factors for a particular plant and their ranges of applicability should be included in the Safety Analysis Report or reload submittal.
Justification Sensitivity factors were evaluated using the WRB-1 and ABB-NV correlations and the VIPRE code for parameter values applicable to the 15x15 DFRA and UPGRADE fuel at EPU conditions. These sensitivity factors were used to determine the RTDP design limit DNBR values. The RTDP design limit DNBR values will be included in the Turkey Point FSAR.
2. "Any changes in DNB correlation, THINC-IV correlations, or parameter values listed in Table 3-1 of WCAP-11397 outside of previously demonstrated acceptable ranges require re-evaluation of the sensitivity factors and of the use of Equation (2-3) of the topical report."
Justification Because the VIPRE code was used to replace the THINC-IV code, sensitivity factors were evaluated for using the VIPRE code. VIPRE has been demonstrated to be equivalent to the THINC-IV code in WCAP-14565-P-A (Reference A.8-1). See the response to condition 3 for a discussion of the use of Equation (2-3) of the topical report. Evaluations using both WRB-1 and ABB-NV correlations were done in compliance with WCAP-11397 methodology.
3. "If the sensitivity factors are changed as a result of correlation changes or changes in the application or use of the THINC code, then the use of an uncertainty allowance for application of Equation (2-3) must be re-evaluated and the linearity assumption made to obtain Equation (2-17) of the topical report must be validated.
Justification Equation (2-3) of WCAP-11397-P-A (Reference A.8-2) and the linearity approximation made to obtain Equation (2-17) were confirmed to be valid for the Turkey Point EPU using the combination of the VIPRE code and the WRB-1 correlation, as well as the ABB-NV correlation.
4. "Variances and distributions for input parameters must be justified on a plant-by-plant basis until generic approval is obtained."
Justification The plant specific variances and distributions were justified for the EPU and are presented in Section 2.8.3.
5. "Nominal initial condition assumptions apply only to DNBR analyses using RTDP. Other analyses, such as overpressure calculations, require the appropriate conservative initial condition assumptions."
Justification Nominal conditions were only applied to the DNBR analyses which used RTDP.

Table A.8-1 (Continued)
Revised Thermal Design Procedure for Non-LOCA Thermal Hydraulics

Limitations, Restrictions, and Conditions
6. "Nominal conditions chosen for use in analyses should bound all permitted methods of plant operation.
Justification Bounding nominal conditions were used in the DNBR analyses using RTDP, consistent with the proposed methods of plant operation for the EPU.
7. "The code uncertainties specified in Table 3-1 (of WCAP-11397-P) (± 4 percent for THINC-IV and ± 1 percent for transients) must be included in the DNBR analyses using RTDP."
Justification The code uncertainties specified in Table 3-1 of WCAP-11397-P-A (Reference A.8-2) remained unchanged and were included in the DNBR analyses using RTDP. The THINC-IV uncertainty was applied to VIPRE, based on the equivalence of the VIPRE model approved in WCAP-14565-P-A to THINC-IV.

References

A.8-1 WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic safety Analysis," Y. X. Sung, et al., October 1999.

A.8-2 WCAP-11397-P-A, "Revised Thermal Design Procedure," Friedland, A. J. and Ray, S., April 1999.

A.9 BEST-ESTIMATE LARGE BREAK LOCA

The following discussion of the applicability limits and usage conditions imposed on the ASTRUM methodology used for the Large Break LOCA analysis is fashioned after the discussion in Section 13-3 of the ASTRUM topical (WCAP-16009-P-A). Only those limits and conditions which have been determined as applicable to the ASTRUM methodology (discussed in Section 13-3 of WCAP-16009-P-A and approved by the NRC in Section 4.0 of the ASTRUM SER) are addressed below.

Table A.9-1
Best-Estimate Large Break LOCA - Applicability Limits

1. "The use of the WCOBRA/TRAC EM for long term cooling licensing analyses is not covered in this review."
The WCOBRA/TRAC EM was used for the Large Break LOCA licensing. The WCOBRA/TRAC thermal-hydraulic computer code was used in the post-LOCA analyses for analyzing the switch to cold leg recirculation. The approach used in the post-LOCA analyses for analyzing the switch to cold leg recirculation is consistent with the method used for the Prairie Island fuel transition: U.S. Nuclear Regulatory Commission, "Prairie Island Nuclear Generating Plant, Units 1 and 2 – Issuance of AMENDMENTS RE: Technical Specifications Changes to Allow Use of Westinghouse 0.422-inch OD 14X14 VANTAGE+ Fuel (TAC Nos. MD9142 and MD9143)," July 1, 2009 (ML091460809). As such, this limit is met.
2. "Our review did not cover the use of the WCOBRA/TRAC EM for small break LOCA licensing analyses."
The WCOBRA/TRAC EM was used for the Large Break LOCA licensing analysis, but not the small break LOCA analysis. As such, this applicability limit is met.
3. "Section 2.4.4 of this SER [for WCAP-14449-P-A] discusses that ranges and biases of parameters were based on data, including UPTF and CCTF data. Of particular concern is the ranging of interfacial drag and condensation, which is based on UPTF and CCTF data. In a letter dated April 8, 1999, to assure that the 2-loop version of the methodology would not be applied for heat generation rates higher than covered by the UPTF and CCTF data, W proposed to limit the application of the UPI methodology to nominal power levels of 1980 MWt, low power region average heat generation rate of less than 6.9 kW/ft, and maximum analyzed linear heat generation rates of 17 kW/ft. We find the proposed limits are acceptable because they are consistent with the range of the UPTF and CCTF data. We also find that the use of the methodology above these values is outside the scope of our review, and would require further justification and NRC review."
Turkey Point does not have UPI and thus this SER requirement is N/A

Table A.9-2
Best-Estimate Large Break LOCA - Usage Conditions

1. "A recommended justification for any future time step changes (first listed item). We require that W perform this justification as recommended, and retain traceable documentation of this action in its in-house plant records."
This requirement is satisfied since all time step changes have been justified and documented in Westinghouse records.
2. "Based on Reference 214 [A.9-1], Attachment 7, the analysis to determine the uncertainty distributions for accumulator and SI temperatures uses plant operating data and/or plant Technical Specifications. Therefore, this analysis must be performed for each plant."
This requirement is satisfied since the analyzed accumulator and SI temperature ranges use plant operating data.
3. "On CQD [A.9-2] page 7-24, Westinghouse stated the fuel pellet thermal expansion model in MATPRO-11, Revision 1, Reference 176 [A.9-3], was simplified by omitting the corrections for molten fuel and mixed oxide (Pu). In Reference 214 [A.9-1], List II, Item 6, Westinghouse committed to resubmitting the relevant WCOBRA/TRAC models for NRC review if the code will be used to analyze US licensed plants with molten fuel or mixed oxide."
This requirement is satisfied since the PTN Large Break LOCA analysis does not support the use of molten fuel or mixed oxide.
4. "Westinghouse, in Reference 214 [A.9-1], List II, Item 8, committed to not changing the value and range of the broken loop cold leg nozzle loss coefficient for plant specific applications. Also, the values developed apply only to LBLOCA and must be justified for other applications."
This requirement is satisfied since the range of the broken loop cold leg nozzle loss coefficient developed for LBLOCA was not changed for the PTN Large Break LOCA analysis.
5. "Westinghouse, in Reference 214 [A.9-1], Attachment 9, gave additional explanation on its use of the full Method of Characteristics model for each time step in the code implementation of choked flow. In the above reference, Westinghouse committed to include the information in the CQD [A.9-2]."
Westinghouse satisfied this requirement by adding the necessary text to the critical flow model description in Section 4-8-2 of WCAP-12945-P-A and the ASTRUM topical report (WCAP-16009-P-A).
6. "Westinghouse noted that the choked flow solution is implemented in the pressure solution of the code rather than in the back substitution step after solving the pressure equation. This results in a smoother pressure and flow response in the code. In Reference 214 [A.9-1], Attachment 9, Westinghouse committed to include this information in the CQD [A.9-2]."
Westinghouse satisfied this requirement by adding the necessary text to the critical flow model description in Section 4-8-2 of WCAP-12495-P-A and the ASTRUM topical report (WCAP-16009-P-A).

Table A.9-2 (Continued)
Best-Estimate Large Break LOCA - Usage Conditions

7. "Westinghouse, in Reference 214 [A.9-1], List II Item 10, committed to use the multiplier given in Reference 214 [A.9-1], Attachment 4, to account for rod-to-rod radiation effects in the heat transfer multiplier data base."
Westinghouse applies a correction factor to the reflood heat transfer multipliers to account for rod-to-rod radiation effects, as described on page 25-5-26 of WCAP-12945-P-A. The same correction factor is applied with ASTRUM.

References

A.9-1 N. J. Liparulo, Westinghouse, letter to USNRC Document Control Desk, "Docketing of Supplemental Information Related to WCAP-12945-P," NSA-SAI-96-156, April 30, 1996.

A.9-2 S. M. Bajorek, et. Al., WCAP-12945-P-A, Volume 1, Revision 2, and Volumes 2 through 5, Revision 1, "Code Qualification Document for Best Estimate LOCA Analysis," 1998.

A.9-3 D. L. Hagrman, G. A. Reymann, and R. E. Manson, MATPRO-Version 11 (Revision 1), A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior, NUREG/CR-0497, Rev. 1, 1980.