St. Lucie Unit 2
Extended Power Uprate
Licensing Report

Attachment 5
Appendix A

Safety Evaluation Report Compliance

This coversheet plus 38 pages

Appendix A

Safety Evaluation Report Compliance

A.1 Safety Evaluation Report Compliance Introduction

This appendix is a summary of NRC-approved codes and methods used in the licensing report (LR) Section 2.8.5 series for the extended power uprate (EPU). The appendix addresses compliance with the limitations, restrictions, and conditions (LRC) specified in the approving safety evaluation of the applicable codes and methods (NRC Review Standard (RS-001, Section 2.1, Matrix 8, Note 7).

LR Table A.1-1 presents an overview of the safety evaluation reports (SER) by codes and methods as documented in the Westinghouse topical reports. For each topical report, LR Table A.1-1 lists the applicable LR sections and Appendix A sections. The Appendix A sections that follow contain a detailed listing and dispositioning of each LRC imposed by the SER, as identified with “Yes” in column 5 of LR Table A.1-1. Those SERs identified with “No” in column 5 are listed in the reference list of each section, but there is no topical report entry in the companion table.

Table A.1-1
Safety Evaluation Report Compliance Summary

No.	Subject	Topical Report (Reference) / Date of NRC Acceptance	Code(s) or Topic	Limitation, Restriction, Condition (LRC)	Licensing Report Section	Appendix Section
1	LBLOCA	CENPD-132 (A.1-1) / June 1975	Original EM	Yes	2.6.6.2 and 2.8.5.6.3.2	A.2
2		S1 to CENPD-132 (A.1-2)/ June 1975	Original EM	Yes		
3		S2 to CENPD-132 (A.1-3)/ December 1975	1975 EM	Yes		
4		S3 to CENPD-132 (A.1-4)/ July 1986	1985 EM	Yes		
5		S4 to CENPD-132 (A.1-5)/ December 2000	1999 EM	Yes		
6		CENPD-133 (A.1-6) / June 1975	CEFLASH-4A	Yes		
7		S2 to CENPD-133 (A.1-7) / June 1975	CEFLASH-4A	Yes		
8		S4 to CENPD-133 (A.1-8) / July 1986	CEFLASH-4A	No		
9		S5 to CENPD-133 (A.1-9) / July 1986	CEFLASH-4A	Yes		
10		CENPD-134 (A.1-10) / June 1975	COMPERC-II	Yes		
11		S1 to CENPD-134 (A.1-11)/ June 1975	COMPERC-II & COMZIRC	Yes		
12		S2 to CENPD-134 (A.1-12)/ July 1986	COMPERC-II	Yes		
13		CENPD-135 (A.1-13)/ June 1975	STRIKIN-II	Yes		

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

No.	Subject	Topical Report (Reference) / Date of NRC Acceptance	Code(s) or Topic	Limitation, Restriction, Condition (LRC)	Licensing Report Section	Appendix Section
14		S2 to CENPD-135 (A.1-14)/ June 1975	STRIKIN-II	Yes		
15		S4 to CENPD-135 (A.1-15)/ November 1976	STRIKIN-II	Yes		
16		S5 to CENPD-135 (A.1-16)/ September 1978	STRIKIN-II	Yes		
17		CENPD-138 (A.1-17)/ June 1975	PARCH	No		
18		S1 to CENPD-138 (A.1-18)/ June 1975	PARCH	No		
19		S2 to CENPD-138 (A.1-19)/ April 1978	PARCH	Yes		
20		CENPD-213 (A.1-20)/ August 1976	FLECHT Geometry Correction	Yes		
21		Enclosure 1 to LD-81-095 (A.1-21) / July 1986	Flow Blockage	Yes		
22		CEN-372 (A.1-22) / April 1990	Fuel RIP	Yes		
23		CENPD-404 (A.1-23) / September 2001	ZIRLO® Clad	Yes		

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

No.	Subject	Topical Report (Reference) / Date of NRC Acceptance	Code(s) or Topic	Limitation, Restriction, Condition (LRC)	Licensing Report Section	Appendix Section
24	SBLOCA	CENPD-137 (A.1-24) / June 1975	Original EM	Yes	2.8.5.6.3.3	A.3
25		S1 to CENPD-137 (A.1-25)/ September 1977	S1M	No		
26		S2 to CENPD-137 (A.1-26)/ December 1997	S2M	No		
27		CENPD-133 (A.1-27) / June 1975	CEFLASH-4A	No		
28		S1 to CENPD-133 (A.1-28)/ June 1975	CEFLASH-4AS	No		
29		S3 to CENPD-133 (A.1-29)/ September 1977	CEFLASH-4AS	No		
30		CENPD-134 (A.1-30)/ June 1975	COMPERC-II	No		
31		S1 to CENPD-134 (A.1-31)/ June 1975	COMPERC-II & COMZIRC	No		
32		S2 to CENPD-134 (A.1-32)/ July 1986	COMPERC-II	No		
33		CENPD-135 (A.1-33)/ June 1975	STRIKIN-II	No		
34		S2 to CENPD-135 (A.1-34)/ June 1975	STRIKIN-II	No		
35		S4 to CENPD-135 (A.1-35)/ November 1976	STRIKIN-II	Yes		
36		S5 to CENPD-135 (A.1-36)/ September 1978	STRIKIN-II	Yes		
37		CENPD-138 (A.1-37) / June 1975	PARCH	No		
38		S1 to CENPD-138 (A.1-38)/ June 1975	PARCH	No		
39		S2 to CENPD-138 (A.1-39)/ April 1978	PARCH	Yes		
40		CENPD-136 (A.1-40) / June 1975	Fuel Properties	No		
41		CENPD-185 (A.1-41) / October 1975	Zircaloy Burst	Yes		
42		CEN-203-P, Revision 1-P-A (A.1-42) / June 1985	Post TMI II.K.3.30 Action Items	Yes		
43		S1 to CEN-203-P (A.1-43) / June 1985		Yes		

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

No.	Subject	Topical Report (Reference) / Date of NRC Acceptance	Code(s) or Topic	Limitation, Restriction, Condition (LRC)	Licensing Report Section	Appendix Section
44	SBLOCA (continued)	S2 to CEN-203-P (A.1-44) / June 1985	Post TMI II.K.3.30 Action Items (continued)	Yes	2.8.5.6.3.3 (continued)	A.3 (continued)
45		S3 to CEN-203-P (A.1-45) / February 1987		No		
46		S4 to CEN-203-P (A.1-46) / February 1987		No		
47		CENPD-404 (A.1-47) / September 2001	ZIRLO® Clad	Yes		
48	Post LOCA Boric Acid Precipitation	CENPD-254-P-A (A.1-48) / July 1979 Suspended in References A.1-49, and A.1-50	BORON	Yes	2.8.5.6.3.5	A.4
49	LOCA Hydraulic Blowdown Loads	CENPD-252-P-A (A.1-51, and A.1-52) / February 1979	CEFLASH-4B	Yes	2.8.5.6.3.4	A.5
50	Realignment Guidelines Procedures	CENPD-254-P-A (A.1-48) / July 1979 Suspended in References A.1-49, and A.1-50	CELDA, NATFLOW, & CEPAC	Yes	2.8.5.6.3.6	A.4
51	Post-LOCA Criticality	CENPD-254-P-A (A.1-48) / July 1979 Suspended in References A.1-49, and A.1-50	BORON	Yes	2.8.5.6.3.7	A.4
52	Non-LOCA Thermal Transients	WCAP-7908-A (Reference A.1-53/ September 30, 1986)	FACTRAN	Yes	2.8.5.4.1 2.8.5.4.6	A.6
53	Non-LOCA Safety Analysis	WCAP-14882-P-A (Reference A.1-54/ 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 2.8.5.2.4 2.8.5.2.5 2.8.5.3.1 2.8.5.3.2 2.8.5.4.2 2.8.5.4.3 2.8.5.5 2.8.5.6.1 2.8.5.6.2	A.7

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

No.	Subject	Topical Report (Reference) / Date of NRC Acceptance	Code(s) or Topic	Limitation, Restriction, Condition (LRC)	Licensing Report Section	Appendix Section
54	Neutron Kinetics	WCAP-7979-P-A (Reference A.1-55/ July 29, 1974)	TWINKLE	None for Non-LOCA Transient Analysis	2.8.5.4.1 2.8.5.4.6	Not Applicable
55	Multi-dimensional Neutronics	WCAP-10965-P-A & WCAP-11596-P-A (Reference A.1-56/ June 23, 1986), (Reference A.1-57/ May 17, 1988)	ANC	None for Non-LOCA Transient Analysis	2.8.5.1.2 2.8.5.4.3	Not Applicable
56	Non-LOCA Thermal / Hydraulics	WCAP-14565-P-A & Addendum 1-A (Reference A.1-58 / January 19, 1999), (Reference A.1-59 / April 14, 2004)	VIPRE	Yes	2.8.5.1.2 2.8.5.3.1 2.8.5.3.2 2.8.5.4.1 2.8.5.4.2 2.8.5.4.3	A.8

References

A.1-1 CENPD-132 P, Volume I, Calculative Methods for the C-E Large Break LOCA Evaluation Model, August 1974.
CENPD-132 P, Volume II, Calculative Methods for the C-E Large Break LOCA Evaluation Model, August 1974.

A.1-2 CENPD-132 P, Supplement 1, Calculational Methods for the C-E Large Break LOCA Evaluation Model, February 1975.

A.1-3 CENPD-132-P, Supplement 2-P, Calculational Methods for the C-E Large Break LOCA Evaluation Model, July 1975.

A.1-4 CENPD-132, Supplement 3-P-A, Calculative Methods for the C-E Large Break LOCA Evaluation Model for the Analysis of C-E and W Designed NSSS, June 1985.

A.1-5 CENPD-132, Supplement 4-P-A, Calculative Methods for the C-E Nuclear Power Large Break LOCA Evaluation Model, March 2001.

A.1-6 CENPD-133P, CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis, August 1974.

A.1-7 CENPD-133P Supplement 2, CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis (Modifications), February 1975.

A.1-8 CENPD-133 Supplement 4-P, CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis, April 1977.

A.1-9 CENPD-133 Supplement 5-A, CEFLASH-4A, A FORTRAN77 Digital Computer Program for Reactor Blowdown Analysis, June 1985.

A.1-10 CENPD-134P, COMPERC-II, A Program for Emergency Refill-Reflood of the Core, August 1974.

A.1-11 CENPD-134P, Supplement 1, COMPERC-II, A Program for Emergency Refill-Reflood of the Core (Modifications), February 1975.

A.1-12 CENPD-134, Supplement 2-A, COMPERC-II, A Program for Emergency Refill-Reflood of the Core, June 1985.

A.1-13 CENPD-135P, STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program, August 1974.

A.1-14 CENPD-135P, Supplement 2, STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program (Modifications), February 1975.

A.1-15 CENPD-135, Supplement 4-P, STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program, August 1976.

A.1-16 CENPD-135-P, Supplement 5, STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program, April 1977.

A.1-17 CENPD-138P, “PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup, August 1974.

A.1-18 CENPD-138, Supplement 1, PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup (Modifications), February 1975.

A.1-19 CENPD-138, Supplement 2-P, PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup, January 1977.

A.1-20 CENPD-213-P, Application of FLECHT Reflood Heat Transfer Coefficients to C-E’s 16×16 Fuel Bundles, January 1976.

A.1-21 Enclosure 1-P-A to LD-81-095, C-E ECCS Evaluation Model Flow Blockage Analysis, December 1981.

A.1-22 CEN-372-P-A, Rev. 000, Fuel Rod Maximum Allowable Gas Pressure, May 1990.

A.1-23 CENPD-404-P-A, Rev. 0, Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs, November 2001.

A.1-24 CENPD-137P, Calculative Methods for the C-E Small Break LOCA Evaluation Model, August 1974.

A.1-25 CENPD-137, Supplement 1-P, Small Break Model, Calculative Methods for the C-E Small Break LOCA Evaluation Model, January 1977.

A.1-26 CENPD-137, Supplement 2-P-A, Calculative Methods for the ABB CE Small Break LOCA Evaluation Model, April 1998.

A.1-27 Same as A.1-6.

A.1-28 CENPD-133P, Supplement 1, CEFLASH-4AS, A Computer Program for the Reactor Blowdown Analysis of the Small Break Loss of Coolant Accident, August 1974.

A.1-29 CENPD-133, Supplement 3-P, CEFLASH-4AS, A Computer Program for the Reactor Blowdown Analysis of the Small Break Loss of Coolant Accident, January 1977.

A.1-30 Same as A.1-10.

A.1-31 Same as A.1-11.

A.1-32 Same as A.1-12.

A.1-33 Same as A.1-13.

A.1-34 Same as A.1-14.

A.1-35 Same as A.1-15.

A.1-36 Same as A.1-16.

A.1-37 Same as A.1-17.

A.1-38 Same as A.1-18.

A.1-39 Same as A.1-19.

A.1-40 CENPD-136 P (proprietary version), CENPD-136, Rev. 01 (non-proprietary version), High Temperature Properties of Zircaloy and UO_2 for Use in LOCA Evaluation Models, July 1974.

A.1-41 CENPD-185-P-A, Clad Rupture Behavior, LOCA Rupture Behavior of 16x16 Zircaloy Cladding, November 1975.

A.1-42 CEN-203-P, Revision 1-P-A, Response to NRC Action Plan Item II.K.3.30 Justification of Small Break LOCA Methods, March 1982.

A.1-43 CEN-203-P, Revision 1-P, Supplement 1-P-A, Response to NRC Request Number 1 for Additional Information on C-E Report CEN-203-P, Rev 1-P (Response to NRC Action Plan Item II.K.3.30, Justification of Small Break LOCA Methods), February 1984.

A.1-44 CEN-203-P, Revision 1-P, Supplement 2-P-A, Further Response to NRC Request Number 1 for Additional Information on C-E Report CEN-203-P, Rev 1-P (Response to NRC Action Plan Item II.K.3.30, Justification of Small Break LOCA Methods), November 1984.

A.1-45 CEN-203-P, Revision 1-P, Supplement 3-A, Post-Test Analysis of Semiscale Test S-UT-8, Response to NRC's Conditional SER Issued June 20, 1985 on the Justification of C-E Small Break LOCA Methods, December 1985.

A.1-46 CEN-203-P, Revision 1-P, Supplement 4-A, Response to NRC Request for Additional Information for Verification of Analysis Methods for Small Break LOCA's, November 1986.

A.1-47 Same as A.1-23.

A.1-48 CENPD-254-P-A, Post-LOCA Long Term Cooling Evaluation Model, June 1980.

A.1-49 NRC letter, Suspension of NRC Approval for use of Westinghouse Topical Report CENPD-254-P, ‘Post-LOCA Long-Term Cooling Model,’ Due to Discovery of Non-Conservative Modeling Assumptions During Calculations Audit, R. A. Gramm, August 1, 2005. (ADAMS No. ML051920310)

A.1-50 NRC letter, Clarification of NRC Letter Dated August 1, 2005, Suspension of NRC Approval for use of Westinghouse Topical Report CENPD-254-P, ‘Post-LOCA Long-Term Cooling Model,’ Due to Discovery of Non-Conservative Modeling Assumptions during Calculations Audit (TAC MB1365), D. S. Collins, November 23, 2005. (ADAMS No. ML053220569)

A.1-51 CENPD-252-P-A, Blowdown Analysis Method - Method for the Analysis of Blowdown Induced Forces in a Reactor Vessel, July, 1979.

A.1-52 R.L. Baer (NRC) to A. E. Scherer (CE), Staff Evaluation of Topical Report CENPD-252-P, February 12, 1979.
R.L. Baer (NRC) to A. E. Scherer (CE), ‘Staff Evaluation of Topical Report CENPD-252-P, February 28, 1979.

A.1-53 WCAP-7908-A, FACTRAN – A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod, H. G. Hargrove, December 1989.

A.1-54 WCAP-14882-P-A (Proprietary) and WCAP-15234-A (Non-Proprietary), RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses, D. S. Huegel, et al., April 1999.

A.1-55 WCAP-7979-P-A, TWINKLE – A Multi-Dimensional Neutron Kinetics Computer Code, D. H. Risher, Jr. and R. F. Barry, January 1975.

A.1-56 WCAP-10965-P-A, ANC - A Westinghouse Advanced Nodal Computer Code, Y. S. Liu, et al., September 1986.

A.1-57 WCAP-11596-P-A, Qualification of the Phoenix – P/ANC Nuclear Design System for Pressurized Water Reactor Cores, June 1988.

A.1-58 WCAP-14565-P-A (Proprietary) and WCAP-15306-NP-A (Non-Proprietary), VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis, Y. X. Sung, et al., October 1999.

A.1-59 WCAP-14565-P-A, Addendum 1-A, Addendum 1 to WCAP-14565-P-A Qualification of ABB Critical Heat Flux Correlations with VIPRE-01 Code, August 2004.

A.2 The Evaluation Model for Large Break LOCA (LBLOCA)

The large break loss-of-coolant accident (LBLOCA) analysis for EPU was performed using the latest, NRC-approved, standard methodology for CE plants documented in numerous topical report submittals, listed as References A.1-1 through A.1-23. These references document the overall LBLOCA methodology as well as the individual computer codes within the array of codes used by the methodology process. The NRC SERs are specifically referenced in LR Table A.2-1 for each topical report. These SERs state a number of limitations, restrictions, and conditions (LRCs) on the use of the computer codes and the methodology process for licensing basis calculations. The following is a detailed listing of these SER LRCs and the compliance for the EPU LBLOCA analysis:

Table A.2-1
LBLOCA

Item No.	Limitations, Restrictions and Conditions (LRCs)	Compliance
CENPD-132 (Ref. A.1-1) and Supplement 1 to CENPD-132 (Ref. A.1-2) SER Ref. A.2-1		
1	The C-E ECCS evaluation model is acceptable for typical current C-E three and four loop plants.	St. Lucie Unit 2 is a C-E four loop plant.
2	The C-E ECCS evaluation model is acceptable for dry containments (including sub-atmospheric).	St. Lucie Unit 2 is a dry containment plant.
3	The C-E ECCS evaluation model is acceptable for power ratings up to 3800 MWt.	St. Lucie Unit 2 has core power less than 3800 MWt.
4	The C-E ECCS evaluation model is acceptable for plants utilizing only bottom flooding ECCS.	St. Lucie Unit 2 utilizes a bottom flooding ECCS design.
5	NRC acceptance applies only to the use of the C-E ECCS evaluation model for ECCS analyses.	This St. Lucie Unit 2 design analysis is an ECCS performance analysis.
6	Empirical equations for temperature of fuel rod cladding rupture and circumferential swelling as a function of differential pressure must be shown to be applicable, or be modified, when cladding other than that used to develop the equations is being analyzed.	The 1985 EM (Ref. A.1-4), which implemented the NUREG-0630 cladding rupture and blockage models, supersedes this LRC for 16x16 fuel assemblies with Zircaloy-4 cladding. The ZIRLO® cladding topical report (Ref. A.1-23) supersedes it for fuel assemblies with ZIRLO® cladding.

Table A.2-1 (Continued)
LBLOCA

Item No.	Limitations, Restrictions and Conditions (LRCs)	Compliance
7	Even though generic results for the worst single failure of ECCS equipment were reviewed by NRC, the specific application of a worst single failure criterion will be confirmed on each submittal. With each application, the NRC will examine the plant configuration to confirm that the appropriate single failure assumptions have been made.	This LRC for the 1985 EM is addressed by the worst single failure parametric study that was performed for this St. Lucie Unit 2 1999 EM (Ref. A.1-5) analysis.
8	Operational characteristics of the containment safety systems along with containment internal heat sinks, and free volume must be provided for NRC review for each plant type analyzed for ECCS performance.	The COMPERC-II input decks and the Comprehensive Checklist support NRC review of the required containment systems.
9	LBLOCA mass and energy release to the containment data must be submitted for NRC review as part of individual plant submittals.	St. Lucie Unit 2 analysis using this methodology was previously approved by the NRC. Representative LBLOCA mass and energy release data for the bounding limiting break case for this analysis is available and can be provided for review, if requested.
10	Reflood heat transfer coefficients for the 16x16 fuel assembly design based on the FLECHT data must be calculated with a 0.8 multiplier.	This LRC is removed by CENPD-213 and its SER (Ref. A.2-7).
	Supplement 2 to CENPD-132 (Ref. A.1-3) SER Ref. A.2-2	
1	Model change proposed to remove 0.8 multiplier for FLECHT reflood heat transfer coefficients for 16x16 fuel assemblies is not approved for use. C-E must continue to use the 0.8 multiplier on the 14x14 FLECHT correlation for 16x16 applications.	This LRC is removed by CENPD-213 and its SER (Ref. A.2-7).
2	NRC acceptance applies only to the use of the topical report as part of the C-E ECCS evaluation model and only to the use for ECCS analyses.	This St. Lucie Unit 2 design analysis is an ECCS performance analysis.

Table A.2-1 (Continued)
LBLOCA

Item No.	Limitations, Restrictions and Conditions (LRCs)	Compliance
3	Model applicable to typical C-E three and four loop plants.	St. Lucie Unit 2 is a C-E four loop plant.
4	Model applicable to dry containments (including subatmospheric).	St. Lucie Unit 2 is a dry containment plant.
5	Model applicable to power ratings up to 3800 MWt.	St. Lucie Unit 2 has core power less than 3800 MWt.
6	Model applicable to plants utilizing only bottom flooding ECCS.	St. Lucie Unit 2 utilizes a bottom flooding ECCS design.
Supplement 3 to CENPD-132 (Ref. A.1-4) SER Ref. A.2-3		
1	Future large break LOCA analyses of C-E NSSS should use an axial power shape similar to Shape B or one as near to this shape as the Axial Shape Index (ASI) allows. This new axial power shape should be used in the break flow sensitivity study discussed above as a condition for acceptance of the revised break nodalization.	The axial shape used in this analysis is similar to Shape B.
2	The C-E large break model is applicable to all C-E designed PWRs being supplied with C-E manufactured Zircaloy clad fuel.	The SER for the 1999 EM (Ref. A.2-9) revises this LRC to be "With the exception noted in this report, the model is applicable to all CE-designed PWRs with Zircaloy clad fuel." St. Lucie Unit 2 is a C-E PWR that uses fuel assemblies, which utilize Zircaloy-4, and ZIRLO® clad fuel. The SER for the ZIRLO® topical report (Ref. A.2-10) found the 1999 EM acceptable for the analysis of ZIRLO® cladding.
3	Should a cladding rupture temperature greater than 950°C be encountered in any future plant analysis, C-E will submit justification for extending their models into this region.	This LRC applies only for Zircaloy cladding. No Zircaloy cladding rupture temperatures greater than 950°C (1742°F) were encountered in this analysis for St. Lucie Unit 2.

Table A.2-1 (Continued)
LBLOCA

Item No.	Limitations, Restrictions and Conditions (LRCs)	Compliance
4	The highest clad temperature resulting from the HCROSS calculation should occur immediately downstream of the blockage and the heat transfer coefficients (HTCs) resulting from HCROSS should be no greater than the heat transfer coefficients that would result from the FLECHT-based correlation.	The highest cladding temperature in the axial region of the hot rod that uses the HCROSS-PARCH HTCs occurs in the node immediately downstream from the rupture node. Also, this analysis does not use HTCs resulting from HCROSS that are greater than HTCs calculated from the FLECHT-based correlation.
5	For each application of this model (COMPERC-II) it must be determined if the highest peak cladding temperature occurs with or without the worst single failure involving the coolant injection pumps.	This LRC is superseded by the 1999 EM (Ref. A.1-5).
CENPD-133 (Ref. A.1-6) and Supplement 2 to CENPD-133 (Ref. A.1-7) SER Ref. A.2-1		
1	The CEFLASH-4A code option for the pseudo viscosity pressure drop term described in Appendix E of CENPD-133P, Supplement 2 is not allowed.	The pseudo viscosity pressure drop term is not used in this analysis.
2	For each case in the analysis, the initial STRIKIN-II and CEFLASH-4A volumetric average fuel temperatures at the maximum power location in the calculation must be shown to be equal to or greater than that calculated in the approved version of FATES.	For STRIKIN-II, a generic study demonstrated that STRIKIN-II matches the stored energy and fuel rod temperature distributions calculated by FATES3B. For CEFLASH-4A, the fuel centerline temperatures that are input via Card Series 5031 and 5041 bound the values calculated by FATES3B. Also in the 1999 EM, the CEFLASH-4A hot assembly is initialized using the STRIKIN-II average rod conditions that are calculated using FATES3B conditions for the time-in-life specified for the case.

Table A.2-1 (Continued)
LBLOCA

Item No.	Limitations, Restrictions and Conditions (LRCs)	Compliance
3	The staff reviewed C-E's approach to the end-of-bypass and finds that: (1) sustained downward flow in the flow path connecting the two downcomer nodes is an acceptable definition of the Time of Annulus Downflow (TAD). In this context, sustained downflow must represent a flow reversal to downward flow that will be maintained for the rest of the blowdown without significant return to upward flow; and (2) the subtraction of a transport time from the time of annulus downflow is acceptable provided the average fluid velocity used in the transport time calculation is conservatively calculated, for example using a heterogeneous model with a slip ratio of 1.0. The documentation should specify the method used to calculate the average fluid velocity.	This St. Lucie Unit 2 analysis selects time of annulus downflow (TAD) as the time when the upper downcomer node packs as a consequence of the inflow of injection from the SITs. Experience has shown that this initiates a period of sustained downward flow from the upper downcomer node to the lower downcomer node for plants, such as St. Lucie Unit 2, that have 600 psi SITs. Calculation of the transport time to define end-of-bypass is not performed for this St. Lucie Unit 2 analysis. For conservatism, TAD is reported instead of an end-of-bypass time for the ECCS analysis.
4	The LBLOCA core flow distribution during blowdown must be analyzed using one assembly for the hot region.	The hot region of the CEFLASH-4A core model is one assembly.
5	To maximize peak cladding temperature, the degree of swelling at rupture must be modeled using the best estimate equation rather than the upper limit equation.	The 1985 EM (Ref. A.1-4), which implemented the NUREG-0630 cladding rupture and blockage models, supersedes this LRC for Zircaloy-4 cladding. The ZIRLO® cladding topical report (Ref. A.1-23) supersedes it for ZIRLO® cladding.
Supplement 5 to CENPD-133 (Ref. A.1-9) SER Ref. A.2-3		
1	Application of CEFLASH-4A to evaluate Westinghouse plants has not been reviewed nor approved for use.	St. Lucie Unit 2 is a C-E PWR.

Table A.2-1 (Continued)
LBLOCA

Item No.	Limitations, Restrictions and Conditions (LRCs)	Compliance
2	The staff requires that future large break LOCA evaluations with CEFLASH-4A confirm that the limiting break flow discharge coefficient has been determined by an appropriate break spectrum.	This St. Lucie Unit 2 analysis includes an appropriate break spectrum study that confirms that the limiting break flow discharge coefficient has been determined.
3	While the homogeneous equilibrium model (HEM) is recognized as predicting more realistic break flow (compared to the Appendix K Moody model), it is not acceptable for use in design basis LOCA evaluation (per requirements of Appendix K to 10 CFR Part 50).	The St. Lucie Unit 2 analysis uses the Henry-Fauske/Moody critical flow model.
CENPD-134 (Ref. A.1-10) and Supplement 1 to CENPD-134 (Ref. A.1-11) SER Ref. A.2-1		
1	The COMPERC-II code option for the use of a loss coefficient for determination of the injection section pressure drop described in Section II.D of CENPD-134P, Supplement 1, is not allowed.	The subject loss coefficient option for the injection section pressure drop is not used in this analysis.
2	The NRC guide for approved methods, which is provided in the SER, must be used in selecting containment initial conditions, active heat removal systems, and passive heat sinks.	This analysis uses containment data that conforms to this LRC.
3	To account for interaction of steam and ECC water injected in the cold legs, the hydrodynamic differential pressure values across 60° and 75° injection sections must correspond to 0.4 and 1.5 psid, respectively.	The value of 0.4 psid for a 60° injection angle is used in this analysis.
4	The methodology for calculating the delay time for ECC injected water to penetrate the annulus due to the force of gravity and due to the presence of hot walls must follow the NRC's prescription given in the SER.	The St. Lucie Unit 2 COMPERC-II analysis used the NRC methodology.

Table A.2-1 (Continued)
LBLOCA

Item No.	Limitations, Restrictions and Conditions (LRCs)	Compliance
Supplement 2 to CENPD-134 (Ref. A.1-12) SER Ref. A.2-3		
1	For each application of this model (COMPERC-II) it must be determined if the highest peak cladding temperature occurs with or without the worst single failure involving the coolant injection pumps.	This LRC is superseded by the 1999 EM (Ref. A.1-5), which is used for the St. Lucie Unit 2 analysis.
CENPD-135 (Ref. A.1-13) and Supplement 2 to CENPD-135 (Ref. A.1-14) SER Ref. A.2-1		
1	A time-in-life sensitivity study is required for each complete analysis to ensure that the worst case conditions for the initial hot rod stored energy are represented; that is, the initial pin temperatures and pressures must result in the highest peak cladding temperature.	This St. Lucie Unit 2 analysis includes a time-in-life sensitivity study that ensures that the time-in-life that is analyzed results in the highest peak cladding temperature.
2	Fuel rod cladding plastic swelling prior to rupture must be modeled in all cases using the model used by the NRC.	The St. Lucie Unit 2 analysis uses the NRC model for fuel rod cladding plastic swelling prior to rupture.
3	The rod-to-rod thermal radiation model for calculating heat transfer on the hot rod must be based on conservative fuel rod peaking factors for the surrounding fuel rods in the radiation enclosure.	This analysis uses conservative fuel rod peaking factors for the surrounding fuel rods in the radiation enclosure. The peaking factors are based on bounding minimum values for the radiation enclosure X-factor and the hot rod pin-to-box factor.

Table A.2-1 (Continued)
LBLOCA

Item No.	Limitations, Restrictions and Conditions (LRCs)	Compliance
4	For each case in the analysis, the initial STRIKIN-II and CEFLASH-4A volumetric average fuel temperatures at the maximum power location in the calculation must be shown to be equal to or greater than that calculated in the approved version of FATES.	For STRIKIN-II, a generic study demonstrated that STRIKIN-II matches the stored energy and fuel rod temperature distributions calculated by FATES3B. For CEFLASH-4A, the fuel centerline temperatures bound the values calculated by FATES3B. Also in the 1999 EM (Ref. A.1-5), the CEFLASH-4A hot assembly is initialized using the STRIKIN-II average rod conditions that are calculated using FATES3B conditions for the time-in-life specified for the case.
5	To maximize peak cladding temperature, the degree of swelling at rupture must be modeled using the best estimate equation rather than the upper limit equation.	The 1985 EM (Ref. A.1-4), which implemented the NUREG-0630 cladding rupture and blockage models, supersedes this LRC for Zircaloy-4 cladding. The ZIRLO® cladding topical report (Ref. A.1-23) supersedes it for ZIRLO® cladding.
Supplement 4 to CENPD-135 (Ref. A.1-15) SER Ref. A.2-4		
1	NRC acceptance of CENPD-135, Supplement 4-P, applies only to its use as part of the C-E ECCS evaluation model and does not constitute acceptance of it for any other purpose.	This analysis is an ECCS performance analysis.

Table A.2-1 (Continued)
LBLOCA

Item No.	Limitations, Restrictions and Conditions (LRCs)	Compliance
2	For ECCS analyses of plants which show a reflood peak cladding temperature of more than 50°F greater than the blowdown peak, STRIKIN-II calculations based on steam entry into the fuel rod gap at the time of cladding rupture are acceptable. For ECCS analyses of plants which show a blowdown peak cladding temperature within 50°F greater than the reflood peak cladding temperature, STRIKIN-II calculations must be reanalyzed using a delayed steam entry model for the fuel rod gap after cladding rupture is calculated to occur.	This analysis uses the appropriate model for steam entry into the fuel rod gap.
Supplement 5 to CENPD-135 (Ref. A.1-16) **SER Ref. A.2-5**		
1	The version of the STRIKIN-II code described by CENPD-135-P, Supplement 5, is an acceptable version that should be used for future licensing calculations.	The Supplement 5 version of STRIKIN-II is superseded by the 1999 EM version of STRIKIN-II.
Supplement 2 to CENPD-138 (Ref. A.1-19) **SER Ref. A.2-6**		
1	The version of the PARCH code described by CENPD-138, Supplement 2-P, is the only version acceptable for future licensing calculations.	The Supplement 2 version of PARCH is superseded by the 1999 EM version of PARCH.
CENPD-213 (Ref. A.1-20) **SER Ref. A.2-7**		
1	An appendix to CENPD-213-P describes the THERM code which has not been reviewed in detail and is not approved for use at this time.	THERM is not used in this analysis.

Table A.2-1 (Continued)
LBLOCA

Item No.	Limitations, Restrictions and Conditions (LRCs)	Compliance
2	NRC acceptance of the geometry correction method of CENPD-213 applies only to its use as part of the C-E ECCS Evaluation Model and does not constitute acceptance of it for any other purpose.	This analysis is an ECCS performance analysis.
Enclosure 1 to LD-81-095 (Ref. A.1-21) SER Ref. A.2-3		
1	Should a cladding rupture temperature greater than 950°C be encountered in any future plant analysis, C-E will submit justification for extending their models into this region.	No Zircaloy cladding rupture temperatures greater than 950°C (1742°F) were encountered in this analysis for St. Lucie Unit 2.
CEN-372 (Ref. A.1-22) SER Ref. A.2-8		
1	Based on the C-E 14x14 and 16x16 fuel design analyses presented in CEN-372-P, we conclude that the critical rod pressure limit proposed by C-E is acceptable for licensing applications. However, those licensees referencing this high pressure topical report are required to provide plant-specific LOCA analyses to determine the impact of maximum calculated rod pressures on cladding rupture timing and peak cladding temperatures, as described in Section 2.3.	Each potentially limiting fuel type considered in this LBLOCA analysis was analyzed over a burnup range extending from beginning of life to the maximum burnup that a fuel rod may be at the PLHGR (i.e., the burnup at the “knee” of the fuel performance radial fall-off curve). The maximum fuel rod initial stored energy occurs at beginning of life and the maximum rod internal pressure occurs at the knee of the radial fall-off curve for a rod operating at the PLHGR. Beyond this point in burnup, the fuel rod power decreases making the fuel cladding transient no longer limiting for the hot rod PCT during LBLOCA ECCS performance analysis. Therefore, the impact of maximum rod pressure is properly accounted for in the analysis for LOCA.
Supplement 4 to CENPD-132 (Ref. A.1-5) SER Ref. A.2-9		
1	The 1999 EM is applicable to LBLOCA licensing applications for C-E designed pressurized water reactors.	St. Lucie Unit 2 is a C-E PWR.

Table A.2-1 (Continued)
LBLOCA

Item No.	Limitations, Restrictions and Conditions (LRCs)	Compliance
2	The use of the 1999 EM AICS without replacement of the Dougall-Rohsenow correlation for the 1985 EM simulation for licensing applications is not NRC reviewed or approved.	This analysis does not use the 1985 EM simulation feature of the 1999 EM. It uses the 1999 EM with the Dougall-Rohsenow correlation replaced by the Condie-Bengston IV correlation.
3	This SER removes a limitation or constraint from the 1986 SER that limited the C-E cladding swelling and rupture models to C-E manufactured fuel. The new SER constraint is that the model is applicable to all C-E designed PWRs with Zircaloy clad fuel.	St. Lucie Unit 2 is a C-E PWR with Zircaloy-4, and ZIRLO® clad fuel. The SER for the ZIRLO® topical report (Ref. A.2-10) found the 1999 EM acceptable for the analysis of ZIRLO® cladding.
4	This SER provides for licensing closure on a previous submittal. Supplement 4 of CENPD-133 is considered part of the 1985 EM.	No comment necessary.
5	Each licensee that uses the 1999 EM must ensure that the choice of the RWT temperature for safety injection and containment spray provides a bounding PCT result for LBLOCA events.	This analysis includes a RWT temperature parametric study that analyzes minimum and maximum RWT temperature. The temperature that produces the higher PCT is used in the calculation of the PCT for this analysis.
6	The 1999 EM will continue to use the 1985 EM specified inputs for the SG secondary side initial pressure (nominal), SG secondary side initial inventory (nominal), and SG tube plugging (maximum).	This analysis uses the specified inputs for SG secondary side initial pressure, SG secondary side initial inventory, and SG tube plugging.
7	SI actuation in the 1999 EM calculation is based on the Safety Injection Actuation Signal (SIAS) plus delay time.	This analysis models the actuation of the SI on a SIAS generated by a low pressurizer pressure signal with a maximum delay time for pump actuation.

Table A.2-1 (Continued)
LBLOCA

Item No.	Limitations, Restrictions and Conditions (LRCs)	Compliance
8	Each applicant referencing this topical report for the 1999 EM must perform a plant-specific, ECCS component, worst single failure assessment, including consideration of the most limiting value of the RWT temperature listed in constraint 5, above.	This analysis includes a plant-specific ECCS component worst single failure parametric study. The parametric study includes consideration of the most limiting value of the RWT temperature by analyzing both the minimum and maximum RWT temperatures.
CENPD-404 (Ref. A.1-23) SER Ref. A.2-10		
1	All the conditions listed in the SEs for all the CENPD (sic) methodologies used for ZIRLO® fuel analysis will continue to be met, except that the use of ZIRLO® cladding in addition to Zircaloy-4 cladding is now approved.	As documented above, all the LRCs from the SEs associated with the 1985 EM (Ref. A.1-4) and 1999 EM (Ref. A.1-5) are met.
2	All CENP methodologies will be used only within the range for which ZIRLO® data was acceptable and for which the verifications discussed in CENPD-404-P and responses to requests for additional information were performed.	The SER for CENPD-404 found the ZIRLO® cladding models acceptable for use in ECCS performance analyses with no specified limit on the temperature range. Therefore, it is concluded that the models were found acceptable up to the 10 CFR 50.46(b)(1) PCT limit of 2200°F. This analysis uses the 1999 EM only within this temperature range.

References

A.2-1 O. D. Parr (NRC) to F. M. Stern (C-E), NRC Staff Review of the Combustion Engineering ECCS Evaluation Model, June 13, 1975.

A.2-2 O. D. Parr (NRC) to A. E. Scherer (C-E), NRC Staff Review of the Proposed Combustion Engineering ECCS Evaluation Model Changes, December 9, 1975.

A.2-3 D. M. Crutchfield (NRC) to A. E. Scherer (C-E), Safety Evaluation of Combustion Engineering ECCS Large Break Evaluation Model and Acceptance for Referencing of Related Licensing Topical Reports, July 31, 1986.

A.2-4 K. Kniel (NRC) to A. E. Scherer (C-E), Combustion Engineering Emergency Core Cooling System Evaluation Model, November 12, 1976.

A.2-5 R. L. Baer (NRC) to A. E. Scherer (C-E), Evaluation of Topical Report CENPD-135 Supplement No. 5, September 6, 1978.

A.2-6 K. Kniel (NRC) to A. E. Scherer (C-E), Evaluation of Topical Report CENPD-138, Supplement 2-P, April 10, 1978.

A.2-7 K. Kniel (NRC) to A. E. Scherer (C-E), August 2, 1976.

A.2-8 A. C. Thadani (NRC) to A. E. Scherer (C-E), Acceptance for Referencing C-E Topical Report CEN-372-P, Fuel Rod Maximum Allowable Gas Pressure (TAC No. 69231), April 10, 1990.

A.2-9 S. A. Richards (NRC) to P. W. Richardson (WEC), Safety Evaluation of Topical Report CENPD-132, Supplement 4 Revision 1, 'Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model' (TAC No. MA5660), December 15, 2000.

A.2-10 S. A. Richards (NRC) to P. W. Richardson (WEC), Safety Evaluation of Topical Report CENPD-404-P, Revision 0, 'Implementation of ZIRLO Material Cladding in CE Nuclear Power Fuel Assembly Designs' (TAC No. MB1035), September 12, 2001.

A.3 The Evaluation Model for Small Break LOCA (SBLOCA)

The SBLOCA analysis for EPU was performed using the latest, NRC-approved, standard methodology for CE plants documented in numerous topical report submittals, listed as References A.1-24 through A.1-47. These references document the overall SBLOCA methodology as well as the individual computer codes within the array of codes used by the methodology process. The NRC SERs are specifically referenced in LR Table A.3-1 for each topical report. These SERs state a number of limitations, restrictions, and conditions (LRCs) on the use of the computer codes and the methodology process for licensing basis calculations. The following is a detailed listing of these SER LRCs and the compliance for the EPU SBLOCA analysis:

Table A.3-1
SBLOCA

Item No.	Limitations, Restrictions and Conditions (LRCs)	Compliance
	CENPD-137 (Ref. A.1-24) SER Ref. A.3-1	
1	C-E ECCS evaluation model is acceptable for all plants satisfying the following plant classifications: Typical current C-E three and four-loop plant.	St. Lucie Unit 2 is a C-E designed four-loop plant.
2	Dry containment (including subatmospheric)	St. Lucie Unit 2 is a dry containment plant.
3	Power rating up to 3800 MWt	St. Lucie Unit 2 has a power level less than 3800 MWt.
4	Plants utilizing only bottom flooding ECCS.	St. Lucie Unit 2 utilizes a bottom flooding ECCS design.
5	NRC acceptance applies only to the use of the C-E ECCS evaluation model for ECCS analyses.	This analysis is an ECCS performance analysis.
6	Empirical equations for temperature of fuel rod cladding rupture and circumferential swelling as a function of differential pressure must be shown to be applicable, or be modified, when cladding other than that used to develop the equations is being analyzed.	For SBLOCA analysis, CENPD-185-P-A (Ref. A.1-41) and its SER (Ref. A.3-7) provide data for the 16x16 fuel design, thus conforming to this constraint.

Table A.3-1 (Continued)
SBLOCA

Item No.	Limitations, Restrictions and Conditions (LRCs)	Compliance
Supplement 4 to CENPD-135 (Ref. A.1-35) SER Ref. A.3-3		
9	NRC acceptance of CENPD-135, Supplement 4-P, applies only to its use as part of the C-E ECCS evaluation model and does not constitute acceptance of it for any other purpose.	This analysis is an ECCS performance analysis.
Supplement 5 to CENPD-135 (Ref. A.1-36) SER Ref. A.3-4		
10	The version of the STRIKIN-II code described by CENPD-135-P, Supplement 5, is an acceptable version that should be used for future licensing calculations.	STRIKIN-II version STR.2.12 used in this analysis is a Supplement 5 version.
Supplement 2 to CENPD-138 (Ref. A.1-39) SER Ref. A.3-6		
12	The version of the PARCH code described by CENPD-138, Supplement 2-P, is the only version acceptable for future licensing calculations.	For SBLOCA, CENPD-138, Supplement 2-P version of PARCH is superseded by the CENPD-137, Supplement 2-P version of PARCH (Ref. A.3-5).
CENPD-185 (Ref. A.1-41) SER Ref. A.3-7		
13	Model application is limited to 16x16 fuel cladding of System-80 design dimensions.	St. Lucie Unit 2 fuel, manufactured by WEC, uses the dimensions of the 16x16 System-80 fuel cladding design.

Table A.3-1 (Continued)
SBLOCA

Item No.	Limitations, Restrictions and Conditions (LRCs)	Compliance
CEN-203 (Ref. A.1-42) and Supplements 1 and 2 to CEN-203 (Refs. A.1-43 and -A.1-44) SER Ref. A.3-8		
15	NRC review found the submittal acceptable pending a confirmatory benchmark analysis to demonstrate good agreement between CEFLASH-4AS and the data from Semiscale test S-UT-08. In a letter from R. W. Wells (CEOG) to C. O. Thomas (NRC), the CEOG committed to submit results of the benchmark analysis by December 31, 1985. The Staff found this commitment acceptable.	This constraint is removed by the NRC in SER supplement dated February 11, 1987 (Ref. A.3-9).
CENPD-404 (Ref. A.1-47) SER Ref. A.3-10		
1	The staff has found that CENPD-404-P, … is acceptable for referencing in licensing applications for CE designed nuclear power plants, …	St. Lucie Unit 2 is a CE designed nuclear power plant.
2	All CENP methodologies will be used only within the range for which ZIRLO® data was acceptable and for which the verifications discussed in CENPD-404-P and responses to requests for additional information were performed.	The methodologies have been used within the range for which ZIRLO® data was acceptable and for which the verifications were performed as discussed in CENPD-404-P (Ref. A.1-47). The ZIRLO® model is used over the accepted range of temperature (PCT < 2200 ^{o}F) and rupture Δp or hoop stress.

References

A.3-1 O.D. Parr (NRC) to F.M. Stern (C-E), NRC Staff Review of the Combustion Engineering ECCS Evaluation Model, June 13, 1975.

A.3-2 K. Kniel (NRC) to A.E. Scherer (C-E), Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P, September 27, 1977.

A.3-3 K. Kniel (NRC) to A. E. Scherer (C-E), Combustion Engineering Emergency Core Cooling System Evaluation Model, November 12, 1976.

A.3-4 R. L. Baer (NRC) to A. E. Scherer (C-E), Evaluation of Topical Report CENPD-135 Supplement No. 5, September 6, 1978.

A.3-5 T. H. Essig (NRC) to I. C. Rickard (ABB), Acceptance for Referencing of the Topical Report CENPD-137(P), Supplement 2, Calculative Methods for the C-E Small Break LOCA Evaluation Model (TAC No. M95687), December 16, 1997.

A.3-6 K. Kniel (NRC) to A.E. Scherer (C-E), Evaluation of Topical Report CENPD-138, Supplement 2-P, April 10, 1978.

A.3-7 O. D. Parr (NRC) to A. E. Scherer (C-E), October 30, 1975.

A.3-8 C. O. Thomas (NRC) to R. W. Wells (CEOG), Conditional Acceptance for Referencing of Licensing Topical Report CEN-203(P) Rev. 1, Response to NRC Action Plan Item II.K.3.30 Justification of Small Break LOCA Methods, June 20, 1985.

A.3-9 D.M. Crutchfield (NRC) to J.K. Gasper (CEOG), Acceptance for Referencing of Licensing Topical Report," February 11, 1987.

A.3-10 S. A. Richards (NRC) to P. W. Richardson (WEC), Safety Evaluation of Topical Report CENPD-404-P, Revision 0, 'Implementation of ZIRLO Material Cladding in CE Nuclear Power Fuel Assembly Designs' (TAC No. MB1035), September 12, 2001.

A.4 BORON for Post LOCA Boric Acid Precipitation & Post LOCA Criticality and Decay Heat Removal Codes for Realignment Guidelines Procedures

The NRC SER can be found in the front of CENPD-254-P-A (Reference A.1-48). This SER stipulates that there are no conditions and limitations on the use of the BORON code or on the decay heat removal codes CELDA, NATFLOW, and CEPAC for licensing basis calculations. However, in a letter dated August 1, 2005 (Reference A.1-49), the NRC identified concerns regarding the CENPD-254-P-A post LOCA long term cooling evaluation model. The letter states the following:

> "Until the NRC staff's concerns are sufficiently resolved, the staff will not approve the use of TR CENPD-254-P for license applications."

Based on discussions with the NRC staff, the issues identified in Reference A.1-49 must be addressed to the staff's satisfaction for any plant change that impacts the post LOCA long term cooling analysis and that requires NRC approval before implementation. This understanding was confirmed by the NRC in a letter dated November 23, 2005 (Reference A.1-50), wherein the following is stated:

> "Until a supplement to TR CENPD-254-P is issued addressing the staff concerns, the following four items will also need to be addressed by licensees on a plant-specific basis in any future submittals regarding post-LOCA LTC."

The four items identified in Reference A.1-50 are listed in LR Table A.4-1 along with Westinghouse statements of compliance.

Table A.4-1
BORON for Post LOCA Boric Acid Precipitation & Post LOCA Criticality and Decay Heat Removal Codes for Realignment Guidelines Procedures

Limitations, Restrictions and Conditions
1. "The mixing volume must be justified; its calculation must account for void fraction."
Compliance The mixing volume for the St. Lucie Unit 2 boric acid precipitation analysis was justified and accounted for void fraction. The mixing volume is the region in the reactor inner vessel wherein boric acid accumulates as a result of borated water injected by the ECCS equipment replacing the unborated water that leaves the mixing volume in the form of steam produced by boiling in the core. For St. Lucie Unit 2 EPU, changes to the mixing volume from the Reference A.1-48 methodology were made consistent with previously NRC accepted methods and were justified by experimental evidence that was conservatively applied. The liquid volume in the mixing volume was calculated by applying the CEFLASH-4AS phase separation model to this region, thereby incorporating void fraction dependence into the boric acid concentration calculation. The phase separation model used in CEFLASH-4AS was previously approved by the staff for computing the mixture level in the core following small break LOCAs. This model was shown to accurately predict the void fraction and the two-phase mixture level in regions experiencing high rates of heat addition following small break LOCAs.

Table A.4-1 (Continued)
BORON for Post LOCA Boric Acid Precipitation & Post LOCA Criticality and Decay Heat Removal Codes for Realignment Guidelines Procedures

Limitations, Restrictions and Conditions
2. "The calculation of the mixing volume must account for the loop pressure drop between the core and the break."
Compliance The calculated mixing volume for the St. Lucie Unit 2 boric acid precipitation analysis did account for the loop pressure drop between the core and the break. The loop pressure drop was conservatively calculated at several time points including early times with the higher steam flow rate and later times with higher boric acid concentration. Frictional losses in the loop pressure drop calculation were increased by roughly 60% for conservatism and the geometric losses were conservatively modeled with a reactor coolant pump locked rotor hydraulic loss coefficient. The upper elevation of the mixing volume was justified by hydrostatic pressure balances that included the conservatively calculated loop pressure drop between the core and the break.
3. "The boric acid solubility limit must be justified, especially if crediting containment pressures greater than 14.7 psia or chemical additives in the sump water."
Compliance The solubility limit for the St. Lucie Unit 2 boric acid precipitation analysis was determined from the NRC accepted model in Reference A.1-48 for a containment pressure of 14.7 psia; therefore, no credit is taken for containment pressures greater than 14.7 psia. Also no credit is taken in the solubility limit from any impact due to chemical additives in the sump water.
4. "A decay heat multiplier of 1.2 must be used for all times if an Appendix K evaluation model is used."
Compliance Decay heat for the St. Lucie Unit 2 boric acid precipitation analysis is represented with the 1973 ANS Standard with a 1.2 multiplier used for all times, which is a conservative treatment of decay heat following shutdown of the reactor.

A.5 CEFLASH-4B for LOCA Hydraulic Blowdown Loads

The Blowdown Loads assessment of expected results was performed using the standard methodology documented in CENPD-252-P-A using the CEFLASH-4B computer code. CENPD-252-P-A (Reference A.1-51) is the topical report for the LOCA hydraulic blowdown loads methodology using the CEFLASH-4B code. Reference A.1-52 is the SER for Reference A.1-51, and may be found in the front of Reference A.1-51. This SER states a number of conditions and limitations on the use of the CEFLASH-4B evaluation model for licensing basis calculations. The following is a review of these SER restrictions and requirements.

Table A.5-1

CEFLASH-4B for LOCA Hydraulic Blowdown Loads

Limitations, Restrictions and Conditions
1. "The CE critical flow model is to be used."
Compliance Standard methodology in Blowdown Loads analyses uses the CE critical flow model that is described in Section 2.1.3 of the Topical Report. Therefore, Westinghouse is in compliance with this restriction.
2. "The break opening schedules, including location, size and time based on the mechanistic break model employed by Combustion Engineering are to be referenced for licensing calculations."
Compliance Standard methodology in Blowdown Loads analyses addresses mechanistically determined pipe breaks. The mechanistic approach is based on non-linear structural analysis techniques and the conservative assumption of instantaneous crack propagation to determine realistic break opening times. Therefore, Westinghouse is in compliance with this restriction.
3. "The Combustion Engineering design model for the annulus representation is to be used for licensing calculations."
Compliance Standard methodology in Blowdown Loads analyses uses the nodalization for annulus representation that is described in the Topical Report. Therefore, Westinghouse is in compliance with this restriction.
4. "The evaluation of the blowdown induced forces following a postulated LOCA is acceptable provided a CEFLASH-4A licensing calculation is performed to obtain the hydraulic input data."
Compliance In the SER wording, "CEFLASH-4A" should read "CEFLASH-4B (see page 2 of SER)." Standard methodology in Blowdown Loads analyses uses the CEFLASH-4B code. Therefore, Westinghouse is in compliance with this restriction.

A.6 FACTRAN for Non-LOCA Thermal Transients

Table A.6-1
FACTRAN for Non-LOCA Thermal Transients

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 St. Lucie Unit 2: Uncontrolled Rod Withdrawal from Subcritical (St. Lucie Unit 2 UFSAR Section 15.4.1) and Control Element Assembly (CEA) Ejection (St. Lucie Unit 2 UFSAR Section 15.4.8). Initial fuel temperatures used as FACTRAN input in the CEA Ejection analysis were calculated using the NRC-approved FATES3B computer code, as described in CENPD-139-P-A (Reference A.6-1), CEN-161(B)-P-A (Reference A.6-2), and CEN-161(B)-P-SUPPL1-P-A (Reference A.6-3). As indicated in the references, the NRC has approved the method of determining uncertainties for fuel temperatures.
2. "Table 2 presents the guidelines used to select initial temperatures."
Justification 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 in Table A.6-1, Section 1, fuel temperatures were used as input to the FACTRAN code in the CEA Ejection analysis for St. Lucie Unit 2. The assumed fuel temperatures, which were calculated using the FATES3B computer code (Reference A.6-1), 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 Departure from Nucleate Boiling (DNB) concerns) and a low constant value is assumed to maximize fuel temperatures. For the CEA 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.6-1 (Continued)
FACTRAN for Non-LOCA Thermal Transients

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 transients analyzed for St. Lucie Unit 2 that utilize FACTRAN (Uncontrolled Rod Withdrawal from Subcritical (St. Lucie Unit 2 UFSAR Section 15.4.1) and CEA Ejection (St. Lucie Unit 2 UFSAR Section 15.4.8)). 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 transients analyzed for St. Lucie Unit 2 that utilize FACTRAN (Uncontrolled Rod Withdrawal from Subcritical (St. Lucie Unit 2 UFSAR Section 15.4.1) and CEA Ejection (St. Lucie Unit 2 UFSAR Section 15.4.8)).
6. "Although time-independent mechanical behaviours (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 Non-LOCA transients that were analyzed with FACTRAN for St. Lucie Unit 2 (Uncontrolled Rod Withdrawal from Subcritical (St. Lucie Unit 2 UFSAR Section 15.4.1) and CEA Ejection (St. Lucie Unit 2 UFSAR Section 15.4.8)) are included in the list of transients provided in Table 1 of the SER. Table 1 of the SER lists the FACTRAN transients for which time-dependent deformation of the cladding is not expected to occur. For the Uncontrolled Rod Withdrawal from Subcritical transient, relatively low cladding temperatures are involved. For the CEA Ejection transient, a high gap heat transfer coefficient is applied to simulate clad collapse onto the fuel pellet. Both transients are short in duration and the gap heat transfer coefficients applied in FACTRAN are consistent with SER Table 2.

Table A.6-1 (Continued)
FACTRAN for Non-LOCA Thermal Transients

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) is 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 CEA Ejection Transient, the LASER-calculated power distributions were used in the FACTRAN analysis of the CEA Ejection transient for St. Lucie Unit 2.

References

A.6-1 CENPD-139-P-A, Fuel Evaluation Model C-E Fuel Evaluation Model Topical Report, July 1974.

A.6-2 CEN-161(B)-P-A, Improvements to Fuel Evaluation Model, August 1989.

A.6-3 CEN-161(B)-P-SUPPL1-P-A, Improvements to Fuel Evaluation Model, January 1992.

A.7 RETRAN for Non-LOCA Safety Analysis

The use of RETRAN for St. Lucie Unit 2 is approved in Reference A.7-1.

Table A.7-1
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 A.7-1) 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."

Table A.7-1
RETRAN for Non-LOCA Safety Analysis

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 or evaluated for St. Lucie Unit 2 using RETRAN 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

Table A.7-1
RETRAN for Non-LOCA Safety Analysis

1. **"The transients and accidents that Westinghouse proposes to analyze with RETRAN are listed in this SER (Table A.7-1) 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."**

- Steam generator tube rupture
- Break in instrument line or other lines from the reactor coolant pressure boundary that penetrate the containment
- Asymmetric steam generator transient (ASGT)

Each transient analyzed for St. Lucie Unit 2 using RETRAN matches one of the transients listed in Table 1 of the SER, with the exception of the "Break In Instrument Line Or Other Lines From The Reactor Coolant Pressure Boundary That Penetrate The Containment" and the "Asymmetric Steam Generator Transient." Additional justification for these exceptions are provided below:

- The Break in Instrument Line or Other Lines from the Reactor Coolant Pressure Boundary that Penetrate the Containment or Primary Line Break event is analyzed to provide mass release input to the analysis of Radiological Consequences. The limiting primary line break event is modeled as a small primary line break outside containment which may result from a break in a letdown line, instrument line, or a sample line. Since the thermal-hydraulic response of the letdown line break is within the range for events analyzed with RETRAN, such as the Steam Generator Tube Rupture event, the use of RETRAN to analyze the Primary Line Break event for St. Lucie Unit 2 is considered acceptable.
- The limiting ASGT is the sudden closure of a main steam isolation value (or loss of load to one SG). Since the thermal-hydraulic response of the ASGT is within the range for events analyzed with RETRAN, such as the loss of condenser, loss of load, or steamline break events, the use of RETRAN to analyze the ASGT event for St. Lucie Unit 2 is considered acceptable. Note the use of RETRAN to analyze the ASGT event was previously approved in Reference A.7-1.

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

St. Lucie Unit 2 consists of a 2x4 loop Combustion Engineering-designed unit which currently uses RETRAN to perform the Non-LOCA safety analyses of record. The RETRAN approval for St. Lucie Unit 2 is included in Reference A.7-1.

Table A.7-1
RETRAN for Non-LOCA Safety Analysis

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 St. Lucie Unit 2 and Westinghouse sources. Assurance that the RETRAN input data is conservative for St. Lucie Unit 2 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 obtained from the responsible St. Lucie Unit 2 and Westinghouse sources. Consistent with the Westinghouse Reload Evaluation Methodology described in WCAP-9272-P-A (Reference A.7-2), the safety analysis input values used in the St. Lucie Unit 2 analyses were selected to conservatively bound the values expected in subsequent operating cycles.

References

A.7-1 NRC Letter to J. A. Stall (FPL), St. Lucie Plant, Unit No. 2 - Issuance of Amendment Regarding Change in Reload Methodology and Increase in Steam Generator Tube Plugging Limit (TAC NO. MC1566), January 31, 2005.

A.7-2 WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, S.L. Davidson, W. R. Kramer, R. J. Sero, F. W. Kramer, July 1985.

A.8 VIPRE for Non-LOCA Thermal/Hydraulics

Table A.8-1
VIPRE for Non-LOCA Thermal / Hydraulics

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."** **Justification** The ABB-NV correlation with a 95/95 correlation limit of 1.13 was used in the DNB analyses for the St. Lucie Unit 2 CE 16x16 fuel. The use of the ABB-NV DNB correlation is based on the same methodologies as the previously approved safety evaluation supporting the WCAP-9272 Reload Methodology and Implementing 30 percent Steam Generator Tube Plugging Limit. (Reference A.8-1). The use of the plant specific hot channel factors and other fuel dependent parameters in the DNB analysis for the St. Lucie Unit 2 CE 16x16 fuel were justified using the same methodologies as for the previously approved safety evaluation supporting the WCAP-9272 Reload Methodology and Implementing 30 percent Steam Generator Tube Plugging Limit (Reference A.8-1).
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 CE 16x16 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.8-2).
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 the ABB-NV correlation. The DNBR limit for ABB-NV is 1.13. Use of other CHF correlations not currently included in VIPRE will require additional justification."** **Justification** As discussed in response to Condition 1, the ABB-NV correlation with a limit of 1.13 was used in the DNB analyses of CE 16x16 fuel for St. Lucie Unit 2. For conditions where ABB-NV is not applicable, the W-3 DNB correlation was used with a limit of 1.30 for above 1000 psia (a limit of 1.45 for below 1000 psia).

Table A.8-1 (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 St. Lucie Unit 2 safety analysis, the usage 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 (Reference A.8-3) 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.8-1 NRC Letter to J. A. Stall (FPL), ST. LUCIE PLANT, UNIT NO. 2 - ISSUANCE OF AMENDMENT REGARDING CHANGE IN RELOAD METHODOLOGY AND INCREASE IN STEAM GENERATOR TUBE PLUGGING LIMIT (TAC NO.MC1566), January 31, 2005.

A.8-2 WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, S.L. Davidson, W. R. Kramer, R. J. Sero, F. W. Kramer, July 1985.

A.8-3 WCAP-14565-P-A (Proprietary) and WCAP-15306-NP-A (Non-Proprietary), VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis, Y. X. Sung, et al., October 1999.