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U. S. Nuclear Regulatory Commission
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Washington, D. C. 20555-0001

Edwin I. Hatch Nuclear Plant
Joseph M. Farley Nuclear Plant
Vogtle Electric Generating Plant
10 CFR 50.46 ECCS Evaluation Model Annual Reports for 2005

Ladies and Gentlemen:

Pursuant to the reporting requirements of 10 CFR 50.46 (a)(3)(ii), Southern Nuclear Operating Company (SNC) is submitting the emergency core cooling system (ECCS) evaluation model annual reports for Hatch Nuclear Plant Units 1 and 2, Farley Nuclear Plant Units 1 and 2, and Vogtle Electric Generating Plant Units 1 and 2.

These annual reports summarize the nature of and estimated effect of any changes or errors in the ECCS models for the period from January 1, 2005 through December 31, 2005.

This letter contains no NRC commitments. If you have any questions, please advise.

Sincerely,

Lewis Sumner

H. L. Sumner, Jr.

HLS/WAS/daj

Enclosures:

1. Edwin I. Hatch Nuclear Plant 10 CFR 50.46 ECCS Evaluation Model Annual Report for 2005
2. Joseph M. Farley Nuclear Plant 10 CFR 50.46 ECCS Evaluation Model Annual Report for 2005
3. Vogtle Electric Generating Plant 10 CFR 50.46 ECCS Evaluation Model Annual Report for 2005

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Enclosure 1

Edwin I. Hatch Nuclear Plant
10 CFR 50.46 ECCS Evaluation Model
Annual Report for 2005

Enclosure 1
Edwin I. Hatch Nuclear Plant
10 CFR 50.46 ECCS Evaluation Model
Annual Report for 2005

BACKGROUND

In accordance with 10 CFR 50.46(a)(3)(ii), this annual report summarizes the nature of and estimated effect of any changes or errors in the emergency core cooling system (ECCS) model for the period from January 1, 2005 through December 31, 2005 for Hatch Nuclear Plant Units 1 and 2.

DISCUSSION

Updated limiting licensing basis peak clad temperatures (PCTs) applicable to Hatch are provided in the following table.

In 2005 Hatch Units 1 and 2 operated with both GE13 and GE14 fuel in their cores. Therefore, the updated licensing basis PCTs are provided for both GE13 and GE14 fuel. The following table begins by listing the baseline ECCS-LOCA evaluations for GE13 fuel (Reference 1) and GE14 fuel (Reference 2).

The next section of the table lists the applicable changes or errors and their estimated effect on PCT that have previously been reported to the NRC (References 3, 4, 5, 6, and 7).

The final section of the table lists those applicable changes or errors and their estimated effect on PCT during the period from January 1, 2005 through December 31, 2005. There have been no GE 10 CFR 50.46 notifications of changes or errors and no SNC changes to the ECCS model to report for 2005.

CONCLUSION

As documented in the following table, the updated Hatch limiting licensing basis PCTs for GE13 and GE14 remain in compliance with 10 CFR 50.46(b)(1), specifically requiring that the limiting licensing basis PCT shall not exceed 2200 °F. As such, there is no need for reanalysis or taking any other actions in accordance with 10 CFR 50.46(a)(3)(ii) because compliance with 10 CFR 50.46(b)(1) has been maintained.

TABLE 1
EDWIN I. HATCH NUCLEAR PLANT
TOTAL RESULTANT PCT (°F)

Report Period	Description of Change or Error in ECCS Evaluation	Estimated PCT Change (°F)				Updated PCT	
		GE13 Fuel		GE14 Fuel		GE13 Fuel	GE14 Fuel
		PCT Change	Absolute Value	PCT Change	Absolute Value		
Baseline Evaluations	SAFER/ GESTR-LOCA Analysis dated March, 1997 (Ref. 1)	N/A	N/A	N/A	N/A	1688	N/A
	SAFER/ GESTR-LOCA Analysis dated March, 2002 (Ref. 2)	N/A	N/A	N/A	N/A	N/A	1820
Previously Reported Changes or Errors	Hatch 50.46 Annual Report for 2000 (Ref. 3)	10	20	N/A	N/A	1698	N/A
	Hatch 50.46 30 Day Report dated 5/21/01 (Ref. 4)	100	100	N/A	N/A	1798	N/A
	Hatch 50.46 Annual Report for 2002 (Ref. 5)	15	15	0	0	1813	1820
	Hatch 50.46 Annual Report for 2003 (Ref. 6)	10	10	-5	5	1823	1815
	Hatch 50.46 Annual Report for 2004 (Ref. 7)	0	0	0	0	1823	1815
2005 Changes or Errors	None to report	N/A	N/A	N/A	N/A	1823	1815

REFERENCES

1. NEDC-32720P, "Hatch Units 1 and 2 SAFER/GESTR Loss-of-Coolant Accident Analysis," dated March 1997.
2. GE-NE-0000-0000-9200-02P, "Hatch Units 1 and 2 ECCS-LOCA Evaluation for GE14," dated March 2002.
3. SNC Letter HL-6028, H. L. Sumner, Jr. to NRC, "Reporting of Changes and Errors in ECCS Evaluation Models," dated January 31, 2001.
4. SNC Letter HL-6090, H. L. Sumner, Jr. to NRC, "Reporting of Changes and Errors in ECCS Evaluation Models," dated May 21, 2001.
5. SNC Letter NL-03-0999, J. B. Beasley, Jr. to NRC, "10 CFR 50.46 ECCS Evaluation Model Annual Reports for 2002," dated June 2, 2003.
6. SNC Letter NL-04-1042, L. M. Stinson to NRC, "10 CFR 50.46 ECCS Evaluation Model Annual Reports for 2003," dated June 29, 2004.
7. SNC Letter NL-05-1050, H. L. Sumner, Jr. to NRC, "10 CFR 50.46 ECCS Evaluation Model Annual Reports for 2004," dated June 25, 2005.

Enclosure 2

Joseph M. Farley Nuclear Plant
10 CFR 50.46 ECCS Evaluation Model
Annual Report for 2005

Enclosure 2
Joseph M. Farley Nuclear Plant
10 CFR 50.46 ECCS Evaluation Model
Annual Report for 2005

BACKGROUND

In accordance with 10 CFR 50.46(a)(3)(ii), this annual report summarizes the nature of and estimated effect of any changes or errors in the emergency core cooling system (ECCS) model for the period from January 1, 2005 through December 31, 2005 for Farley Nuclear Plant (FNP) Units 1 and 2.

DISCUSSION

The following presents an assessment of the effects of errors and changes to the Westinghouse ECCS Evaluation Models on the FNP Units 1 and 2 loss of coolant accident (LOCA) analysis results since the 2004 annual report (Reference 1) for the calendar year 2005. This annual report has been prepared in accordance with the Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting (WCAP-13451, Reference 2), with the exception of plant changes. Starting in 2001, a change in the Westinghouse reporting methodology was made to include the 50.59 Plant Change PCT values as a part of the 50 °F error reporting section. The 2005 annual report (contained herein) is consistent with the change implemented in the 2001 annual report.

Unit 2 implemented the Reactor Internals Upflow Conversion Program (Reference 3) in 2002, and as such a new PCT rack-up reflecting the new upflow configuration analysis is presented here for Unit 2.

Large-Break LOCA

Table 1A shows the LBLOCA PCT rack-ups for both Unit 1 and Unit 2 for Reflood 1 (Reference 4). Table 1B shows the corresponding large-break LOCA PCT rack-ups for Reflood 2 (Reference 4).

<u>LBLOCA ECCS MODEL ANALYSIS-OF-RECORD</u>

The large-break LOCA analyses for Farley Units 1 and 2 were examined to assess the effects of the changes and errors in the Westinghouse large-break LOCA ECCS Evaluation Model on PCT results.

The large-break LOCA analysis-of-record results for Farley Units 1 and 2 were calculated using Westinghouse's BE-LOCA analysis (References 1 and 4). The Unit 1 and Unit 2 analyses assumed the following information important to the large-break LOCA in the BE-LOCA analysis (References 1 and 4). One analysis was used to bound both Farley Unit 1 and Unit 2.

- 17x17 VANTAGE+ Fuel Assembly
- Core Power = 2775 MWT
- Steam Generator Plugging Level = 20%
- $F_Q = 2.50$
- $F\Delta H = 1.70$

For Farley Units 1 and 2, the limiting size break analysis-of-record is a split break of the cold leg piping with a discharge coefficient of $C_D = 1.0$.

PRIOR LBLOCA ECCS MODEL ASSESSMENTS

Prior 10 CFR 50.46 Assessments Reported as Significant

The following LBLOCA 10 CFR 50.46 assessments were reported in September 2005 as significant (the cumulative change exceeded 50°F and certain individual changes also exceeded 50°F).

Accumulator line/Pressurizer Surge Line Data

It was determined that the design and actual plant accumulator line piping schedule were not the same. A Farley specific BE-LBLOCA sensitivity analysis resulted in a 41 °F benefit for the first Reflood and a 9 °F benefit for the second Reflood when actual plant data was modeled (Reference 7). This assessment is applicable to Unit 1 and Unit 2.

Decay Heat Uncertainty error in Monte Carlo Calculation

It was determined that an error existed in the calculation of decay heat uncertainty in the Monte Carlo calculation of the 95th percentile PCT for BE-LBLOCA (Reference 9). This caused an 8 °F penalty for Unit 1 and 2 on Reflood 1 only.

Revised Blowdown Heatup Uncertainty Distribution

Correction of modeling inconsistencies and input errors in the LOFT input decks have resulted in a change in the predicted peak cladding temperature transients. The overall code uncertainty for blowdown was recalculated and programmed into a new version of MONTECF. This resulted in a 5 °F penalty for Unit 1 and 2 for both the first and second Refloods.

PAD 4.0 Fuel Data

PAD 4.0 fuel data was used in evaluation of RHR pump surveillance testing. Use of the PAD 4.0 fuel data reduces the initial stored energy, thereby resulting in a PCT benefit during Reflood. The PCT benefit of PAD 4.0 fuel data was determined to be 50 °F for Reflood 1 and 65 °F for Reflood 2 for Unit 1 and 2.

RHR Test Configuration SI Flow Reduction

During surveillance testing of the RHR pumps, there would be a reduction in calculated SI flow should a LOCA occur while in the testing alignment. The PCT effects were determined to be negligible for Reflood 1 and a 100 °F penalty for Reflood 2.

Prior 10 CFR 50.59 Assessments

The following two plant change assessments were reported in the last submittal (Reference 1) and occurred prior to 2001.

The addition of permanent storage boxes in containment was evaluated and found not to cause a change to PCT (Reference 6).

The finalization of Replacement Steam Generator Data was evaluated and found not to cause a change to PCT (Reference 1).

CURRENT LBLOCA ECCS MODEL ASSESSMENTS

The following changes and errors in the Westinghouse ECCS Evaluation Model would affect the BE-LOCA Model.

Prior 10 CFR 50.46 Reported Assessments

None

2005 10 CFR 50.46 PCT Assessments

None

CURRENT PLANNED PLANT CHANGE EVALUATIONS

Starting with the 2001 annual report, the 10 CFR 50.59 Plant Change PCT values have been considered to be a part of the 50 °F error reporting section. The 2005 annual report (contained herein) is consistent with the changes implemented in the 2001 annual report.

Prior 10 CFR 50.59 Model Assessments

None

2005 Planned Plant Changes

None

TOTAL RESULTANT LBLOCA PCT

As discussed above, the changes and errors to the Westinghouse large-break LOCA ECCS Evaluation Model could affect the large-break LOCA analysis results by altering the PCT. As shown in Table 1A and Table 1B, the large-break LOCA analysis PCT results for both units are below the 10 CFR 50.46 limit of 2200 °F.

Small-Break LOCA

Table 2 shows the small-break LOCA PCT rack-ups for both Unit 1 and Unit 2.

SBLOCA ECCS MODEL ANALYSIS-OF-RECORD

The small-break LOCA analyses for Farley Units 1 and 2 were also examined to assess the effects of the changes and errors to the Westinghouse small-break LOCA ECCS Evaluation Models on PCT results. The small-break LOCA ECCS analysis results were calculated using the NOTRUMP small-break LOCA ECCS Evaluation Model (Reference 5). As noted earlier, the Unit 2 re-analysis reflects the Reactor Internals Upflow Conversion implemented in 2002 (Reference 3).

The Unit 1 and Unit 2 analyses assumed the following information important to the small-break LOCA analyses:

- o 17x17 VANTAGE+ Fuel Assembly
- o Core Power = 1.02 * 2775 MWT
- o Upflow Configuration
- o F_Q = 2.50
- o $F\Delta H$ = 1.70

For Farley Units 1 and 2, the limiting size break analysis-of-record for the VANTAGE+ fuel analysis is a 3-inch diameter break in the cold leg. The limiting PCT values determined for the Unit 1 and Unit 2 17x17 VANTAGE+ small-break are shown in Table 2.

PRIOR SLBLOCA ECCS MODEL ASSESSMENTS

Prior 10 CFR 50.46 Assessments Reported as Significant

The following SBLOCA 10 CFR 50.46 assessment was reported in March 2000 as significant.

An overall PCT benefit of 62 °F for Unit 1 for the "Burst and Blockage/Time in Life" penalty resulted from the SPIKE computer code correlation revision (Reference 11).

Prior 10 CFR 50.59 Assessments

The following three plant change assessments were reported in the last submittal (Reference 1) and occurred prior to 2001.

The addition of permanent storage boxes in containment was evaluated and found not to cause a change to PCT (Reference 6).

The finalization of Replacement Steam Generator Data resulted in a 62 °F benefit for Unit 1 (Reference 10).

Annular pellets were determined to have a 10 °F penalty for SBLOCA results for Unit 1 (Reference 8).

Note that the Unit 2 result (in Table 2) is unaffected by these prior 50.59 plant changes. The reason is that the Unit 2 Upflow Conversion implemented in 2002 required a small-break LOCA re-analysis that included the above changes explicitly.

CURRENT SBLOCA ECCS MODEL ASSESSMENTS

The following changes and errors were identified:

Prior 10 CFR 50.46 Reported Assessments

The following assessments were reported in the last PCT submittal (Reference 1).

NOTRUMP Mixture Level Tracking/Region Depletion Errors

Several closely related errors have been discovered in how NOTRUMP deals with the stack mixture level transition across a node boundary in a stack of fluid nodes. As previously reported, the impact of this revision on the SBLOCA results has been determined to be a 13 °F penalty for Unit 1. In addition, the associated change in Burst and Blockage/Time in Life Components was an additional 12 °F for Unit 1. Thus, the total change was 25 °F for Unit 1. This error does not impact Unit 2's re-analysis result (see previously discussed Reactor Internals Upflow Conversion), since the re-analysis was performed with the corrected version of NOTRUMP.

2005 10 CFR 50.46 PCT Assessments

None

CURRENT PLANNED PLANT CHANGE EVALUATIONS

Starting with the 2001 annual report, the 10 CFR 50.59 Plant Change PCT values have been considered to be a part of the 50 °F error reporting section. The 2005 annual report (contained herein) is consistent with the change implemented in the 2001 annual report.

Prior 10 CFR 50.59 Model Assessments

None

2005 Planned Plant Changes

None

TOTAL RESULTANT SBLOCA PCT

As discussed above, the changes and errors in the Westinghouse small-break LOCA ECCS Evaluation Model could affect the small-break LOCA analysis results by altering the PCT. As shown in Table 2, the small-break LOCA analysis PCT results for both units are below the 10 CFR 50.46 limit of 2200 °F.

CONCLUSION

As documented in the following tables, the updated Farley large-break and small-break LOCA analyses PCTs remain in compliance with 10 CFR 50.46(b)(1), specifically requiring that the PCT shall not exceed 2200 °F. As such, there is no need for reanalysis or taking any other actions in accordance with 10 CFR 50.46(a)(3)(ii) because compliance with 10 CFR 50.46(b)(1) has been maintained. However, as a separate initiative, SNC has performed reanalysis of the large-break LOCA PCT using the ASTRUM methodology. This new analysis was approved by the NRC for Farley in Amendments 174 for Unit 1 and 167 for Unit 2 issued on July 11, 2006. This change to the analysis of record will be reflected in the 10 CFR 50.46 ECCS Evaluation Model Annual Report for 2006 which will be submitted in 2007.

TABLE 1A
JOSEPH M. FARLEY NUCLEAR PLANT
TOTAL RESULTANT LARGE-BREAK LOCA PCT (°F) FOR REFLOOD 1

	UNIT 1	UNIT 2
A. LBLOCA ECCS MODEL ANALYSIS-OF-RECORD		
1. ECCS Analysis	2056*	2056*
2. Increased Containment Spray Flow	9*	9*
Total Analysis-of-Record	2065*	2065*
B. PRIOR LBLOCA ECCS MODEL ASSESSMENTS		
1. Prior 10 CFR 50.46 Assessments Reported as Significant		
A. Accumulator Line/Pressurizer Surge Line Data	-41*	-41*
B. MONTECF Decay Heat Uncertainty Error	8*	8*
C. Revised Blowdown Heatup Uncertainty Distribution	5#	5#
D. PAD 4.0 Fuel Data	-50**	-50**
E. RHR Test Configuration SI Flow Reduction (note 1)	0**	0**
2. Prior 10 CFR 50.59 Assessments		
A. Addition of Permanent Storage Boxes in Containment	0	0
B. Finalization of Replacement Steam Generator Data	0	0
Sum of Prior Assessments	-78	-78
C. CURRENT LBLOCA ECCS MODEL ASSESSMENTS		
1. None	0	0
D. CURRENT PLANNED PLANT CHANGE EVALUATIONS		
1. None	0	0
E. TOTAL RESULTANT LBLOCA PCT		
Total	**1987#**	**1987#**

The PCT values are rounded up to the next highest integer number to avoid reporting in decimal points. The Analysis of Record PCT results reflect the Replacement Steam Generators analysis values.

* See References 1 and 4
\# See Reference 4
** See Reference 12

Note 1 – Assessment applies during quarterly RHR Pump Testing Configuration only.

TABLE 1B
JOSEPH M. FARLEY NUCLEAR PLANT
TOTAL RESULTANT LARGE-BREAK LOCA PCT (°F) FOR REFLOOD 2

	UNIT 1	UNIT 2
A. LBLOCA ECCS MODEL ANALYSIS-OF-RECORD		
1. ECCS Analysis	1956*	1956*
2. Increased Containment Spray Flow	1*	1*
Total Analysis-of-Record	1957*	1957*
B. PRIOR LBLOCA ECCS MODEL ASSESSMENTS		
1. Prior 10 CFR 50.46 Assessments Reported as Significant		
A. Accumulator Line/Pressurizer Surge Line Data	-9*	-9*
B. MONTECF Decay Heat Uncertainty Error	0*	0*
C. Revised Blowdown Heatup Uncertainty Distribution	5#	5#
D. PAD 4.0 Fuel Data	-65**	-65**
E. RHR Test Configuration SI Flow Reduction (note 1)	100**	100**
2. Prior 10 CFR 50.59 Assessments		
A. Addition of Permanent Storage Boxes in Containment	0	0
B. Finalization of Replacement Steam Generator Data	0	0
Sum of Prior Assessments	31	31
C. CURRENT LBLOCA ECCS MODEL ASSESSMENTS		
1. None	0	0
D. CURRENT PLANNED PLANT CHANGE EVALUATIONS		
1. None	0	0
E. TOTAL RESULTANT LBLOCA PCT		
Total	**1988#**	**1988#**

The PCT values are rounded up to the next highest integer number to avoid reporting in decimal points. The Analysis of Record PCT results reflect the Replacement Steam Generators analysis values.

* See References 1 and 4
# See Reference 4
** See Reference 12

Note 1 – Assessment applies during quarterly RHR Pump Testing Configuration only.

TABLE 2
JOSEPH M. FARLEY NUCLEAR PLANT
TOTAL RESULTANT SMALL-BREAK LOCA PCT (°F)

	UNIT 1	UNIT 2
A. SBLOCA ECCS MODEL ANALYSIS-OF-RECORD		
1. ECCS Analysis	1883*	1868**
2. Burst and Blockage / Time in Life	137*	120**
Total Analysis-of-Record	2020*	1988*
B. PRIOR SBLOCA ECCS MODEL ASSESSMENTS		
1. Prior 10 CFR 50.46 Assessments Reported as Significant	-62*	0
2. Prior 10 CFR 50.59 Assessments		
A. Addition of Permanent Storage Boxes in Containment	0*	0
B. Finalization of Replacement Steam Generator Data	-62#	0
C. Annular Pellet Blanket	10*	0
Sum of Prior Assessments	-114*	0
C. CURRENT SBLOCA ECCS MODEL ASSESSMENTS		
1. NOTRUMP Mixture Level Tracking / Region Depl Errors	13*	**
2. Associated change in Burst and Blockage	12*	**
D. CURRENT PLANNED PLANT CHANGE EVALUATIONS		
1. None	0	0
E. TOTAL RESULTANT SBLOCA PCT		
Total	**1931***	**1988****

The PCT values are rounded up to the next highest integer number to avoid reporting in decimal points.

* See References 1 and 4

** The revised analysis-of-record reflects the Unit 2's conversion of downflow to upflow configuration (see References 1 and 3).

\# See Reference 10

REFERENCES

1. Letter from L. M. Stinson to USNRC (NL-05-1551), "Joseph M. Farley Nuclear Plant, 10 CFR 50.46 Annual ECCS Evaluation Model Report for 2004 and Significant Error Report," September 7, 2005.

2. WCAP-13451, "Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting," October 1992.

3. ALA-02-039, "Transmittal of Reactor Internals Upflow Conversion Program Engineering Report, J. M. Farley Nuclear Plant Unit 2," June 2002 (also see WCAP-15974, November 2002).

4. LTR-LIS-06-117, "10 CFR 50.46 Annual Notification and Reporting for 2005," March 6, 2006.

5. "Westinghouse Small-break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary), WCAP-10081-A (Non-Proprietary), Lee, N., et. al, August 1985.

6. SECL-97-062. Rev. 1, "Effects on LOCA PCT of Adding Permanent Storage Boxes and Lead Blankets Inside Containment," October 17, 1997.

7. ALA-00-037, "Final 10 CFR 50.46 Annual Notification and Reporting," March 8, 2000.

8. WCAP-15098, "Joseph M. Farley Nuclear Plant Units 1 and 2 RSG Program NSSS Licensing Report," November 1998.

9. ALA-01-008, "10 CFR 50.46 Annual Notification and Reporting for 2000," March 6, 2001.

10. ALA-01-01, "Southern Nuclear Operating Company, Joseph M. Farley Nuclear Plant Units 1 and 2, LBLOCA and SBLOCA Impacts Due to Final RSG Data for SGRP," February 11, 2000.

11. Letter from D. N. Morey to USNRC (NEL-00-0080), "Joseph M. Farley Nuclear Plant 10 CFR 50.46 Annual ECCS Evaluation Model Changes Report for 1999 and Significant Error Reports," March 29, 2000.

12. ALA-05-55, "Southern Nuclear Operating Company, Joseph M. Farley Nuclear Plant Units 1 and 2, Transmittal of Quarterly RHR Pump Testing Evaluation Revision 1," July 11, 2005

Enclosure 3

Vogtle Electric Generating Plant
10 CFR 50.46 ECCS Evaluation Model
Annual Report for 2005

Enclosure 3
Vogtle Electric Generating Plant
10 CFR 50.46 ECCS Evaluation Model
Annual Report for 2005

BACKGROUND

In accordance with 10 CFR 50.46(a)(3)(ii), this annual report summarizes the nature of and estimated effect of any changes or errors in the emergency core cooling system (ECCS) model for the period from January 1, 2005 through December 31, 2005 for Vogtle Electric Generating Plant Units 1 and 2.

DISCUSSION

The following presents an assessment of the effects of errors and changes to the Westinghouse ECCS Evaluation Models on the Vogtle Electric Generating Plant (VEGP) Units 1 and 2 loss of coolant accident (LOCA) analysis results since the 2004 annual report (Reference 10) for the calendar year 2005. This annual report has been prepared in accordance with the Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting (WCAP-13451, Reference 1), with the exception of plant changes. Starting in 2001, a change in the Westinghouse reporting methodology was made to include the 50.59 Plant Change PCT values as a part of the 50 °F error reporting section. The 2005 annual report (contained herein) is consistent with the change implemented in the 2001 annual report.

Large-Break LOCA

Table 1A shows the LBLOCA PCT rack-ups for Unit 1. Table 1B shows the LBLOCA PCT rack-ups for Unit 2.

LBLOCA ECCS MODEL ANALYSIS-OF-RECORD

The large-break LOCA analyses for Vogtle Units 1 and 2 were examined to assess the effects of the changes and errors in the Westinghouse large-break LOCA ECCS Evaluation Model on PCT results.

In the Annual Report submitted on June 25, 2005 (Reference 10), SNC reported a LBLOCA PCT of 2040.5 °F for both Unit 1 and Unit 2. This value is based on fuel designs containing 128 Integral Fuel Burnable Absorber (IFBA) rods. During 2005 refueling outages, SNC implemented fuel designs containing 156 IFBA rods for both Unit 1 and Unit 2. SNC maintains separate analyses-of-record for both fuel designs. The LBLOCA PCT for fuel designs with 156 IFBA rods is 2062.1 °F for both Unit 1 and Unit 2.

The large-break LOCA analysis was performed with the 1981 Version of the Westinghouse ECCS Evaluation Model using BASH (Reference 3) including changes in the methodology for execution of the model described in References 4 and 5, and the latest acceptable LOCBART model. The VEGP Unit 1 and Unit 2 analyses assumed the following information important to the large-break LOCA analyses:

- 17x17 VANTAGE+ Fuel Assembly
- Core Power = 1.02 * 3565 MWt
- Vessel Average Temperature = 570.7 °F
- Steam Generator Plugging Level = 10%
- $F_Q = 2.50$
- $F\Delta H = 1.65$

For VEGP Units 1 and 2, the limiting size break continues to be the double-ended guillotine rupture of the cold leg piping with a discharge coefficient of $C_D = 0.6$. The LBLOCA LOCBART analysis-of-record calculated PCT value is 2062.1 °F for both Unit 1 and Unit 2.

PRIOR LBLOCA ECCS MODEL ASSESSMENTS

Prior 10 CFR 50.46 Assessments Reported as Significant

None

Prior 10 CFR 50.59 Assessments

None

CURRENT LBLOCA ECCS MODEL ASSESSMENTS

Prior 10 CFR 50.46 Reported Assessments

None

2005 10 CFR 50.46 PCT Assessments

None

Enclosure 3
10 CFR 50.46 ECCS Evaluation Model Annual Report for 2005

CURRENT PLANNED PLANT CHANGE EVALUATIONS

Prior 10 CFR 50.59 Model Assessments

None

2005 Planned Plant Changes

None

TOTAL RESULTANT LBLOCA PCT

For Unit 1, the absolute sum of the LBLOCA PCT assessments is 0 °F.

For Unit 2, the absolute sum of the LBLOCA PCT assessments is 0 °F.

Small-Break LOCA

Table 2A shows the small-break LOCA PCT rack-ups for Unit 1. Table 2B shows the small-break LOCA PCT rack-ups for Unit 2.

SBLOCA ECCS MODEL ANALYSIS-OF-RECORD

In the Annual Report submitted on June 25, 2005 (Reference 10), SNC reported a SBLOCA PCT of 1138.0 °F for both Unit 1 and Unit 2.

The small-break LOCA analysis was performed with the Westinghouse ECCS Evaluation Model using NOTRUMP (References 6 and 7), including changes to the methodology described in References 8 and 9, and the latest acceptable SBLOCTA model. The VEGP Unit 1 and Unit 2 analyses assumed the following information important to the small-break LOCA analyses:

- 17x17 VANTAGE+ Fuel Assembly
- Core Power = 1.02 * 3565 MWt
- Vessel Average Temperature = 570.7 °F
- Steam Generator Plugging Level = 10%
- F_Q = 2.58
- $F\Delta H$ = 1.70

For VEGP Units 1 and 2, the limiting size small-break continues to be a three-inch equivalent diameter break in the cold leg. The SBLOCA SBLOCTA analysis-of-record calculated PCT value is 1138.0 °F for both Unit 1 and Unit 2.

PRIOR SLBLOCA ECCS MODEL ASSESSMENTS

Prior 10 CFR 50.46 Assessments Reported as Significant

None

Prior 10 CFR 50.59 Assessments

None

CURRENT SBLOCA ECCS MODEL ASSESSMENTS

Prior 10 CFR 50.46 Reported Assessments

None

2005 10 CFR 50.46 PCT Assessments

None

CURRENT PLANNED PLANT CHANGE EVALUATIONS

Prior 10 CFR 50.59 Model Assessments

None

2005 Planned Plant Changes

None

TOTAL RESULTANT SBLOCA PCT

For Unit 1, the absolute sum of the SBLOCA PCT assessments is 0 °F.

For Unit 2, the absolute sum of the SBLOCA PCT assessments is 0 °F.

CONCLUSION
As documented in the following tables, the updated VEGP large-break and small-break LOCA analyses PCTs remain in compliance with 10 CFR 50.46(b)(1), specifically requiring that the PCT shall not exceed 2200 °F. As such, there is no need for reanalysis or taking any other actions in accordance with 10 CFR 50.46(a)(3)(ii) because compliance with 10 CFR 50.46(b)(1) has been maintained.

TABLE 1A
VOGTLE ELECTRIC GENERATING PLANT
TOTAL RESULTANT LARGE-BREAK LOCA PCT (°F) FOR UNIT 1

Based on the preceding discussions concerning the VEGP-specific application of the Westinghouse BASH large-break ECCS Evaluation Model, the licensing basis LBLOCA PCT is as follows:

A.	**LBLOCA ECCS MODEL ANALYSIS-OF-RECORD**	
	1. LOCBART Analysis Result (156 IFBA)	2062.1 °F
B.	**PRIOR LBLOCA ECCS MODEL ASSESSMENTS**	
	1. Combined assessments previously reported as significant	+0 °F
	2. Combined planned plant change evaluations	+0 °F
C.	**CURRENT LBLOCA ECCS MODEL ASSESSMENTS**	
	1. None	+0 °F
D.	**CURRENT PLANNED PLANT CHANGE EVALUATIONS**	
	1. None	+0 °F
E.	**TOTAL RESULTANT LBLOCA PCT**	
	Total	**2062.1 °F**

TABLE 1B
VOGTLE ELECTRIC GENERATING PLANT
TOTAL RESULTANT LARGE-BREAK LOCA PCT (°F) FOR UNIT 2

Based on the preceding discussions concerning the VEGP-specific application of the Westinghouse BASH large-break ECCS Evaluation Model, the licensing basis LBLOCA PCT is as follows:

A.	**LBLOCA ECCS MODEL ANALYSIS-OF-RECORD**	
	1. LOCBART Analysis Result (156 IFBA)	2062.1 °F
B.	**PRIOR LBLOCA ECCS MODEL ASSESSMENTS**	
	1. Combined assessments previously reported as significant	+0 °F
	2. Combined planned plant change evaluations	+0 °F
C.	**CURRENT LBLOCA ECCS MODEL ASSESSMENTS**	
	1. None	+0 °F
D.	**CURRENT PLANNED PLANT CHANGE EVALUATIONS**	
	1. None	+0 °F
E.	**TOTAL RESULTANT LBLOCA PCT**	
	Total	**2062.1 °F**

TABLE 2A
VOGTLE ELECTRIC GENERATING PLANT
TOTAL RESULTANT SMALL-BREAK LOCA PCT (°F) FOR UNIT 1

Based on the preceding discussions concerning the VEGP-specific application of the Westinghouse NOTRUMP small-break ECCS Evaluation Model, the licensing basis SBLOCA PCT is as follows:

A. SBLOCA ECCS MODEL ANALYSIS-OF-RECORD	
1. SBLOCTA Analysis Result	1138.0 °F
B. PRIOR SBLOCA ECCS MODEL ASSESSMENTS	
1. Combined assessments previously reported as significant	+0 °F
2. Combined planned plant change evaluations	+0 °F
C. CURRENT SBLOCA ECCS MODEL ASSESSMENTS	
1. None	+0 °F
D. CURRENT PLANNED PLANT CHANGE EVALUATIONS	
1. None	+0 °F
E. TOTAL RESULTANT SBLOCA PCT	
Total	**1138.0 °F**

TABLE 2B
VOGTLE ELECTRIC GENERATING PLANT
TOTAL RESULTANT SMALL-BREAK LOCA PCT (°F) FOR UNIT 2

Based on the preceding discussions concerning the VEGP-specific application of the Westinghouse NOTRUMP small-break ECCS Evaluation Model, the licensing basis SBLOCA PCT is as follows:

A. SBLOCA ECCS MODEL ANALYSIS-OF-RECORD	
1. SBLOCTA Analysis Result	1138.0 °F
B. PRIOR SBLOCA ECCS MODEL ASSESSMENTS	
1. Combined assessments previously reported as significant	+0 °F
2. Combined planned plant change evaluations	+0 °F
C. CURRENT SBLOCA ECCS MODEL ASSESSMENTS	
1. None	+0 °F
D. CURRENT PLANNED PLANT CHANGE EVALUATIONS	
1. None	+0 °F
E. TOTAL RESULTANT SBLOCA PCT	
Total	**1138.0 °F**

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3. "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," WCAP-10266-P-A, Revision 2 (Proprietary) and WCAP-11524-A, Revision 2 (Non-Proprietary), March 1987.

4. Westinghouse letter NTD-NRC-94-4143 from N. J. Liparulo to W. T. Russell (USNRC), "Change in Methodology for Execution of BASH Evaluation Model," May 23, 1994.

5. Westinghouse letter NTD-NRC-95-4540 from N. J. Liparulo to W. T. Russell (USNRC), "Change in Methodology for Execution of BASH Evaluation Model," August 29, 1995.

6. "NOTRUMP: A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A (Proprietary) and WCAP-10080-A (Non-Proprietary), August 1985.

7. "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A, August 1985.

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

9. "Model Changes to the Westinghouse Appendix K Small Break LOCA NOTRUMP Evaluation Model: 1988 – 1997," WCAP-15085, July 1998.

10. NL-05-1050, "10 CFR 50.46 ECCS Evaluation Model Annual Reports for 2004," (multi-docket) letter from H. L. Sumner, Jr. (SNC) to USNRC, June 25, 2005.