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W3F1-2014-0024

10 CFR 50.46

March 25, 2014

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

SUBJECT: 10 CFR 50.46 Significant Change Report for the Waterford 3 Emergency Core Cooling System Performance Analysis Due to Implementation of the Replacement Steam Generators
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

REFERENCES
1. CENPD-132, Supplement 4-P-A, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," March 2001.
2. CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model," April 1998.

Dear Sir or Madam:

The analyses for the LBLOCA and SBLOCA ECCS performance were re-analyzed for the implementation of Waterford 3 Replacement Steam Generators (RSG). The RSGs were installed in refueling outage (RF18), and the analyses became effective after entry into Mode 1 on January 17, 2013 at 14:02. Contrary to the requirements of 10 CFR 50.46(a)(3)(ii) for reporting, a 30-day report that had been prepared was not forwarded to the NRC. This has been entered into the Waterford Steam Electric Station, Unit 3 (Waterford 3) Corrective Action Program in condition report CR-WF3-2014-00955.

This letter is submitted pursuant to 10 CFR 50.46(a)(3)(ii) to provide notification of a significant change to the peak cladding temperature of the Large Break Loss-of-Coolant Accident (LBLOCA) and Small Break Loss-of-Coolant Accident (SBLOCA) Emergency Core Cooling System (ECCS) performance analyses for Waterford 3.

As noted above, the analyses for the LBLOCA and SBLOCA ECCS performance had been re-analyzed for the implementation of Waterford 3 Replacement Steam Generators (RSG). In addition, the new analyses incorporated an allowance for up to 10% steam generator tube plugging. The re-analyses were performed using the latest NRC accepted versions of the Westinghouse evaluation models for Combustion Engineering designed pressurized water reactors (References 1 and 2). The basis for this report is not associated with any model or calculational errors.

The results of the new LBLOCA analysis conforms to the ECCS acceptance criteria of 10 CFR 50.46(b) as discussed in the attachment to this letter. Because the sums of the absolute magnitudes of the changes in peak cladding temperature (PCT) associated with the changes implemented in the new analyses are greater than 50°F (reduction), the changes qualify as significant as defined in 10 CFR 50.46 (a)(3)(i). The PCT change for the SBLOCA analysis does not specifically meet the requirements of 10 CFR 50.46(b) for reporting as a significant change. However, since it results in a PCT that approaches this criterion, the change in PCT for the SBLOCA analysis is also being reported. As noted above, these analyses became effective after entry into Mode 1 from the RSG refueling outage (RF18) which was completed in January 2013. The details of these analyses were incorporated into the latest revision of the Waterford 3 Updated Final Safety Analysis Report in compliance with 10 CFR 50.71(e).

This letter contains no new commitments. If you have any questions or require additional information, please contact me at 504-739-6685.

Sincerely,

JPJ/RJP

Attachment: 10 CFR 50.46 Significant Change Report for Changes to the Waterford 3 ECCS Performance Analysis

cc: Mr. Mark L. Dapas, Regional Administrator
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Attachment to

W3F1-2014-0024

10 CFR 50.46 Significant Change Report for Changes to the Waterford 3 ECCS Performance Analysis

INTRODUCTION

This significant change report is provided for Waterford 3 in accordance with the requirements of 10 CFR 50.46(a)(3)(ii) (Reference 1) for reporting:

(1) changes in an acceptable evaluation model or the application of such a model that affects the temperature calculation and

(2) the estimated effect of the changes on the limiting Emergency Core Cooling System (ECCS) analysis.

Because the effects on the Peak Clad Temperature (PCT) of the changes described herein are greater than 50°F, the changes qualify as significant as defined in 10 CFR 50.46(a)(3)(i) and, consequently, are provided in this significant change report. No errors are being reported.

Emergency Core Cooling System performance for the Large Break Loss of Coolant Accident (LBLOCA) and the Small Break Loss of Coolant Accident (SBLOCA) has been re-analyzed for Waterford 3. The analyses were performed with the latest NRC-accepted versions of the Westinghouse Appendix K evaluation models for Combustion Engineering designed Pressurized Water Reactors (PWRs). The analyses modeled the implementation of Replacement Steam Generators (RSG) with allowance for up to 10% steam generator tube plugging (SGTP).

The new LBLOCA and SBLOCA analyses are not assessments (i.e., they do not provide an estimate of the effect of the changes on the limiting ECCS analysis). Rather, they are complete re-analyses that use acceptable evaluation models that are applicable to Waterford 3. A summary description of the new analyses and their compliance with 10 CFR 50.46 is provided below.

LBLOCA ECCS PERFORMANCE ANALYSIS

LBLOCA Evaluation Model

The new LBLOCA ECCS performance analysis was performed with the 1999 Evaluation Model (EM) version of the Westinghouse LBLOCA evaluation model for Combustion Engineering designed PWRs (Reference 3), an Appendix K EM (Reference 2). Additionally, the analysis used methodology supplements for the implementation of ZIRLO™ and Optimized ZIRLO™ cladding types, the implementation of CE 16x16 Next Generation Fuel (NGF) assemblies, and the use of the 1999 EM Optional Steam Cooling Model. Each of these 1999 EM submittals has been generically accepted by the NRC for licensing applications for Combustion Engineering designed PWRs. These elements of the LBLOCA EM are described in the subsections below.

In accordance with the requirements of Waterford 3 Technical Specification 6.9.1.11, the 1999 EM topical reports are listed in Section III of the Core Operating Limits Report (COLR) as approved analytical methodologies that can be used to determine core operating limits in the COLR.

The new LBLOCA analysis complies with the limitations/constraints imposed by the Safety Evaluation Reports (SERs) for the 1999 EM topical reports as well as the applicable limitations/constraints imposed by the SERs for earlier versions of the LBLOCA evaluation model.

1999 EM (CENPD-132, Supplement 4-P-A)

The Westinghouse Appendix K Evaluation Model for ECCS Performance in CE plants is the 1999 Evaluation Model (1999 EM) for LBLOCA (Reference 3). The SERs documenting NRC acceptance of the evaluation model are provided in References 5, 6, 7, and 23. The 1999 EM for LBLOCA is augmented by CENPD-404-P-A for analysis of ZIRLO™ cladding (Reference 13 and approved by NRC in Reference 24), and by Addendum 1 to CENPD-404-P-A for analysis of Optimized ZIRLO™ cladding (Reference 14 and approved by NRC in Reference 19). Also, the 1999 EM is supplemented by WCAP-16072-P-A (Reference 15 and approved by NRC in Reference 25) for implementation of ZrB_2 IFBA fuel assembly designs. The 1999 EM includes an optional steam cooling heat transfer component model for less than 1 in/sec core reflood that includes spacer grid heat transfer effects as documented in Reference 12 and approved by NRC in Reference 4. The implementation of CE 16x16 NGF into the 1999 EM methodology is documented in Reference 11 and approved by NRC in Reference 16.

CE 16x16 NGF (WCAP-16500-P)

The methodologies for licensing CE 16x16 NGF assemblies are documented in Westinghouse Topical Report WCAP-16500-P, titled "CE 16 x 16 Next Generation Fuel Core Reference Report" (Reference 11). WCAP-16500-P was approved by the NRC in Reference 16. Section 5.2 of Reference 11 documents the ECCS performance methods suitable for use to analyze the implementation of NGF. The final SER for WCAP-16500-P contains 10 Limitations and Conditions. Compliance with these Limitations and Conditions for implementation of NGF in Waterford 3 was documented in Reference 17, and the ECCS Performance analysis was documented in Reference 18.

Optimized ZIRLO™ (CENPD-404-P-A Addendum 1)

The CE 16x16 NGF design utilizes Optimized ZIRLO™, an advanced cladding alloy. The implementation of Optimized ZIRLO™ in CE plants is documented in Reference 14 and approved by the NRC in Reference 19. As required by the SER Limitations and Conditions in Reference 19, the ECCS performance analysis computer codes have been updated to include the Optimized ZIRLO™ cladding property changes detailed in the topical report. The use of Optimized ZIRLO™ cladding in the NGF assemblies requires a cladding exemption from the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K, which was submitted to the NRC in Reference 20 and accepted by NRC in Reference 21.

1999 EM Optional Steam Cooling Model (CENPD-132 Supplement 4-P-A Addendum 1-P)

"Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model, Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less Than 1 in/sec Core Reflood," CENPD-132, Supplement 4-P-A, Addendum 1-P was approved by the NRC

in Reference 4. For the implementation of RSGs at Waterford 3, the LBLOCA analysis credited the use of the modified model including spacer grid heat transfer effects.

The conclusion of the SER stated that Limitations and Conditions 3, 4, and 5 are applicable to CE 16 x 16 NGF design fuel assemblies. The first two Limitations and Conditions included in the SER for CENPD-132, Supplement 4-P-A, Addendum 1-P are for fuel designs other than CE 16 x 16 NGF. The applicable Limitations and Conditions and the means of satisfying them were documented in Reference 17, and the supplementary ECCS Performance analysis was documented in Reference 22.

Fuel Design Changes

For ECCS Performance modeling using the 1999 EM, there were no fuel design changes implemented for the new analysis with RSGs.

Plant Parameter Changes

The new LBLOCA analysis includes the design impact changes for RSGs. The RSGs were implemented assuming an allowance for up to 10% SGTP in the analysis inputs. Replacement of the Waterford 3 RSGs leads to a significant reduction (i.e., greater than 50 °F) in predicted PCT.

Results and Conclusion of the New LBLOCA Analysis

The new LBLOCA analysis analyzed a break spectrum of four reactor coolant pump discharge leg breaks ranging in size from a full double-ended break to a 0.4 double-ended break. The analysis included a study to determine the most limiting single failure of ECCS equipment. The study analyzed no failure, failure of an emergency diesel generator, and failure of a low pressure and a high pressure safety injection pump. The analysis also included studies that investigated the impact of variations in initial safety injection tank conditions and refueling water tank temperature on PCT and maximum cladding oxidation. The analysis also included studies that investigated the limiting initial fuel rod conditions by performing burnup dependent calculations for both UO_2 and ZrB_2 burnable absorber fuel rods for a full core of CE 16x16 NGF assemblies.

Table 1 compares several important inputs used in the current and the new LBLOCA analyses. Table 2 compares important results from the two analyses. A more detailed description of the new analysis, including tables and figures that present the results of the break spectrum analysis as well as results confirming use of the optional steam cooling model, has been incorporated into the Waterford 3 Updated Final Safety Analysis Report in accordance with 10 CFR 50.71(e).

As shown in Table 2, the net change in PCT that resulted from the new LBLOCA analysis implementing RSGs is a minus 74°F. Also shown in Table 2, the net change in maximum local cladding oxidation percentage from the new LBLOCA analysis implementing RSGs is a minus 3.9%.

As summarized below, the results of the new LBLOCA analysis conform to the acceptance criteria of 10 CFR 50.46(b).

Parameter	Criterion	Result
Peak Cladding Temperature	≤ 2200°F	2092°F
Maximum Cladding Oxidation	≤ 17 %	13.0 %
Maximum Core-Wide Oxidation	≤ 1 %	<1 %
Coolable Geometry	Yes	Yes

The results are applicable to Waterford 3 with RSGs and up to 10% steam generator tubes plugged and for operation at a Peak Linear Heat Generation Rate (PLHGR) of 12.9 kW/ft and a core power of 3735 MWt (rated core power of 3716 MWt with a 0.5% power measurement uncertainty).

The new LBLOCA analysis uses the 1999 EM, which is accepted by the NRC for licensing applications for Combustion Engineering designed PWRs such as Waterford 3. The analysis complies with the limitations/constraints imposed by all applicable SERs. The analysis uses values for plant design data that are either applicable to or bound the current configuration of Waterford 3. Entergy and Westinghouse have ongoing processes that ensure that the as operated plant values for PCT-sensitive parameters remain bounded by the values used in the analysis. The new analysis will be used as the reference analysis to evaluate the impact on PCT of future changes to or errors in the 1999 EM and its application to Waterford 3.

SBLOCA ECCS PERFORMANCE ANALYSIS

SBLOCA Evaluation Model

The small break LOCA analysis used the Supplement 2 version (referred to as the S2M or Supplement 2 Model) of the Westinghouse small break LOCA NRC-accepted Evaluation Model (Reference 8) for Combustion Engineering designed plants. The SERs documenting NRC acceptance of the evaluation model are contained in References 5, 9, and 10. The methodology for modeling the NGF assembly design in ECCS Performance Analyses using the S2M is contained in the CE 16x16 NGF Core Topical Report (Reference 11). This report was approved by the NRC and contains the SER constraints and limitations on the application of the S2M for the analysis of NGF. The S2M for SBLOCA is augmented by CENPD-404-P-A for analysis of ZIRLO™ cladding (Reference 13) and by Addendum 1 to CENPD-404-P-A for analysis of Optimized ZIRLO™ cladding (Reference 14). Also, the S2M is supplemented by WCAP-16072-P-A for implementation of ZrB_2 IFBA fuel assembly designs (Reference 15).

In accordance with the requirements of Waterford 3 Technical Specification 6.9.1.11, the S2M topical reports are listed in Section III of the COLR as approved analytical methodologies that can be used to determine core operating limits in the COLR.

The new SBLOCA analysis complies with the limitations/constraints imposed by the SERs for the S2M topical reports as well as the applicable limitations/constraints imposed by the SERs for earlier versions of the SBLOCA evaluation model.

Fuel Design Changes

For ECCS Performance modeling using the S2M, there were no fuel design changes implemented for the new analysis with RSGs.

Plant Parameter Changes

The new SBLOCA analysis includes the design impact changes for RSGs. The RSGs were implemented assuming an allowance for up to 10% SGTP in the analysis inputs.

Results and Conclusion of the New SBLOCA Analysis

The new SBLOCA analysis analyzed a break spectrum of three reactor coolant pump discharge leg breaks, namely, 0.04 ft^2, 0.05 ft^2, and 0.06 ft^2. The 0.05 ft^2/PD break was determined to be the limiting small break LOCA. The analysis was performed using the failure of a direct current (DC) bus as the most limiting single failure of the ECCS. A DC bus failure would prevent startup of an emergency diesel generator that would cause the loss of a High Pressure Safety Injection (HPSI) pump and a low pressure safety injection (LPSI) pump, and results in a minimum of safety injection water being available to cool the core. The analysis credits operation of the steam generator atmospheric dump valves (ADVs). The ADVs are safety grade equipment. They are modeled in automatic mode with an opening pressure of 1040 psia. The most limiting single failure of a DC bus, which prevents start up of a diesel generator, results in loss of DC power to an ADV controller. Thus only one of the two ADVs (one ADV per SG) is available for control of secondary side pressure. The SBLOCA analysis was performed for the fuel rod conditions that result in the maximum initial stored energy in the fuel. The calculations included the analysis of both UO_2 and ZrB_2 burnable absorber fuel rods for a full core of NGF assemblies.

Table 3 compares several important inputs used in the current and the new SBLOCA analyses. Table 4 compares important results from the two analyses. A more detailed description of the new analysis, including tables and figures that present the results of the break spectrum analysis, have been incorporated into the Waterford 3 Updated Final Safety Analysis Report in accordance with 10 CFR 50.71(e).

As shown in Table 4, the net change in PCT that resulted from the new SBLOCA analysis implementing RSGs is minus 48 °F. Also shown in Table 4, the net change in maximum local cladding oxidation percentage from the new SBLOCA analysis implementing RSGs is minus 3.1%.

As summarized below, the results of the new SBLOCA analysis conform to the acceptance criteria of 10 CFR 50.46(b).

Parameter	Criterion	Result
Peak Cladding Temperature	≤ 2200°F	1925°F
Maximum Cladding Oxidation	≤ 17 %	11.2 %
Maximum Core-Wide Oxidation	≤ 1 %	<0.65 %
Coolable Geometry	Yes	Yes

The results are applicable to Waterford 3 with RSGs and up to 10% steam generator tubes plugged and for operation at a Peak Linear Heat Generation Rate (PLHGR) of 13.2 kW/ft and a core power of 3735 MWt (rated core power of 3716 MWt with a 0.5% power measurement uncertainty).

The new SBLOCA analysis uses the S2M, which is accepted by the NRC for licensing applications for CE designed PWRs such as Waterford 3. The analysis complies with the limitations/constraints imposed by all applicable SERs. The analysis uses values for plant design data that are either applicable to or bound the current configuration of Waterford 3. Entergy and Westinghouse have ongoing processes that ensure that the as operated plant values for PCT-sensitive parameters remain bounded by the values used in the analysis. The new analysis will be used as the reference analysis to evaluate the impact on PCT of future changes to or errors in the S2M and its application to Waterford 3.

Summary of Compliance with 10 CFR 50.46

The new LBLOCA and SBLOCA analyses comply with 10 CFR 50.46 as follows:

- The analyses were performed with acceptable evaluation models and included sensitivity studies that assured the limiting LBLOCA and SBLOCA were analyzed [10 CFR 50.46(a)(1)(i)].
- The results of the new LBLOCA and SBLOCA analyses conform to the ECCS acceptance criteria [10 CFR 50.46(b)].
- This report provides NRC with notification of the change in the application of the evaluation models and their effect on the limiting ECCS analyses [10 CFR 50.46(a)(3)(ii)].

The new LBLOCA and SBLOCA analyses for the implementation of RSGs with allowance for up to 10% SGTP constitute new licensing basis analyses (analyses-of-record) for Waterford 3. These analyses became effective after entry into Mode 1 from the RSG refueling outage (RF18) which was completed in January 2013. The new analyses will be used as the reference analyses to evaluate the impact on PCT of future changes to or errors in the 1999 EM and the S2M and their application to Waterford 3.

REFERENCES

1. Code of Federal Regulations, Title 10, Part 50, Section 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors".
2. Code of Federal Regulations, Title 10, Part 50, Appendix K, "ECCS Evaluation Models".
3. CENPD-132P, "Calculative Methods for the C-E Large Break LOCA Evaluation Model," August 1974.

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

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

   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.

   CENPD-132, Supplement 4-P-A, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," March 2001.
4. NRC Letter to Westinghouse dated June 27, 2007, "Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR) CENPD-132 Supplement 4-P-A, Addendum 1-P, 'Calculative Methods for the CE [Combustion Engineering] Nuclear Power Large Break LOCA Evaluation Model – Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less than 1 in/sec Core Reflood'
5. O.D. Parr (NRC) to F.M. Stern (C-E), June 13, 1975.
6. O.D. Parr (NRC) to A.E. Scherer (C-E), December 9, 1975.
7. 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.
8. CENPD-137P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model," August 1974.

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

   CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model," April 1998.
9. 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.
10. 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.
11. WCAP-16500-P-A, Rev. 0, "CE 16x16 Next Generation Fuel Core Reference Report," August 2007.
12. CENPD-132-P-A Supplement 4-P-A Addendum 1-P-A, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model, Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less Than 1 in/sec Core Reflood," August 2007.

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

14. WCAP-12610-P-A & CENPD-404-P-A Addendum 1-A, "Addendum 1 to WCAP-12610-P-A and CENPD-404-P-A Optimized ZIRLO™," July 2006.

15. WCAP-16072-P-A, Rev. 0, "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs," August 2004.

16. Letter from H. K. Nieh (NRC) to J. A. Gresham (Westinghouse), "Final Safety Evaluation for Westinghouse Electric Company Topical Report WCAP-16500-P, Revision 0, 'CE 16x16 Next Generation Fuel Core Reference Report' (TAC No. MD0560)," July 30, 2007.

17. Entergy letter to the NRC, "License Amendment Request NPF-38-271 to Support Next Generation Fuel," August 2, 2007 (W3F1-2007-0037). (ADAMS No.: ML072180042)

18. Entergy letter to the NRC, "Emergency Core Cooling System Performance Analysis to Support Next Generation Fuel," August 9, 2007 (W3F1-2007-0038). (ADAMS No.: ML072250389)

19. Letter from H. N. Berkow (NRC) to J. A. Gresham (Westinghouse), "Final Safety Evaluation for Addendum 1 to Topical Report WCAP-12610-P-A and CENPD-404-P-A, 'Optimized ZIRLO™' (TAC No. MB8041)," June 10, 2005.

20. Entergy letter to the NRC, "License Amendment Request to Allow the Use of Optimized ZIRLO™ Fuel Rod Cladding," April 24, 2007 (W3F1-2007-0020). (ADAMS No.: ML071160348)

21. Letter from N. Kalyanam (NRC) to Vice President, Operations (EOI), "Waterford Steam Electric Station, Unit 3 – Exemption from Specific Requirements in 10 CFR 50.46 and from Appendix K to 10 CFR Part 50, to Allow the Use of Optimized ZIRLO™ Fuel Rod Cladding Material (TAC No. MD5426)," March 11, 2008. (ADAMS No.: ML080380002)

22. Entergy letter to the NRC, "Supplement to the ECCS Performance Analysis Submitted in Support of Next Generation Fuel – 1999 EM Optional Steam Cooling Model Justification," October 4, 2007 (W3F1-2007-0045). (ADAMS No.: ML072820400)

23. Letter from Mr. S. A. Richards (NRC) to Mr. P. W. Richardson (Westinghouse) dated December 15, 2000, "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)."

24. Letter from Mr. S. A. Richards (NRC) to Mr. P. W. Richardson (Westinghouse) dated September 12, 2001, "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)."

25. Letter from Mr. H. N. Berkow (NRC) to Mr. J. A. Gresham (Westinghouse), dated May 6, 2004, "Final Safety Evaluation for Topical Report WCAP-16072-P, Revision 00, 'Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs,' (TAC No. MB8721)."

Table 1

Waterford 3 LBLOCA ECCS Performance Analysis
Comparison of Important Input Parameters

Parameter	Current Analysis[(a)]	New Analysis
LBLOCA Evaluation Model	1999 EM	1999 EM
Core Power Level, MWt (including power measurement uncertainty)	3735	3735
Peak Linear Heat Generation Rate (PLHGR) of the Hot Rod, kW/ft	12.9	12.9
PLHGR of the Average Rod in Assembly with Hot Rod, kW/ft	12.0	12.0
Reactor Coolant System Flow Rate, lbm/hr	148.0×10^6	148.0×10^6
Core Flow Rate, lbm/hr	144.15×10^6	144.15×10^6
RCS Pressure, psia	2250	2250
Cold Leg Temperature, °F	533	533
Hot Leg Temperature, °F	598.7	598.7
Number of Plugged Tubes per Steam Generator	1870 (OSG)	897 (RSG)
Fuel Assembly Design	CE 16x16 NGF	CE 16x16 NGF
Fuel Rod Cladding Type	Optimized ZIRLO™	Optimized ZIRLO™

(a) Applicable to Cycle 16 and later for Original Steam Generators (OSGs)

Table 2

Waterford 3 LBLOCA ECCS Performance Analysis
Comparison of Important Results

Parameter	**Current Analysis[a]**	**New Analysis[b]**
Limiting Cases for Peak Cladding Temperature		
Limiting Break Size	1.0 DEG/PD[c]	1.0 DEG/PD
Peak Cladding Temperature, °F	2166	2092
Maximum Cladding Oxidation, %	16.9	12.8
Maximum Core-Wide Cladding Oxidation, %	<1.0	<1.0
Limiting Cases for Maximum Cladding Oxidation		
Limiting Break Size	1.0 DEG/PD[c]	0.8 DEG/PD
Peak Cladding Temperature, °F	2155	2091
Maximum Cladding Oxidation, %	16.9	13.0
Maximum Core-Wide Cladding Oxidation, %	<1.0	<1.0

(a) OSGs with 1870 plugged tubes per steam generator
(b) RSGs with 897 plugged tubes per steam generator
(c) DEG/PD = Double-Ended Guillotine Break in Pump Discharge Leg

Table 3

Waterford 3 SBLOCA ECCS Performance Analysis
Comparison of Important Input Parameters

Parameter	Current Analysis[(a)]	New Analysis
SBLOCA Evaluation Model	S2M	S2M
Core Power Level, MWt (including power measurement uncertainty)	3735	3735
Peak Linear Heat Generation Rate, kW/ft	13.2	13.2
RCS Flow Rate, lbm/hr	148.0×10^6	148.0×10^6
Core Flow Rate, lbm/hr	144.15×10^6	144.15×10^6
RCS Pressure, psia	2250	2250
Cold Leg Temperature, °F	552	552
Hot Leg Temperature, °F	615.5	615.5
Number of Plugged Tubes per Steam Generator	1870 (OSG)	897 (RSG)
Fuel Assembly Design	CE 16x16 NGF	CE 16x16 NGF
Fuel Rod Cladding Type	Optimized ZIRLO™	Optimized ZIRLO™

(a) Applicable to Cycle 16 and later for Original Steam Generators

Table 4

Waterford 3 SBLOCA ECCS Performance Analysis
Comparison of Important Results

Parameter	Current Analysis[(a)]	New Analysis[(b)]
Limiting Break Size, ft^2	0.055	0.05
Peak Cladding Temperature, °F	1973	1925
Maximum Cladding Oxidation, %	14.3	11.2
Maximum Core-Wide Cladding Oxidation, %	<0.80	<0.65

(a) OSGs with 1870 plugged tubes per steam generator
(b) RSGs with 897 plugged tubes per steam generator