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Waterford 3

W3F1-2002-0106

December 16, 2002

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

SUBJECT: Waterford Steam Electric Station, Unit 3
Docket No. 50-382
Waterford 3 Extended Power Uprate Methodologies

REFERENCE: NRC Letter dated October 25, 2002, "Summary of Meeting Held on October 16, 2002, With Entergy Operations, Inc. (EOI), Re: Extended Power Uprate at Waterford Steam Electric Station, Unit 3 (Waterford 3)"

Dear Sir or Madam:

Entergy Operations, Inc. (Entergy) appreciated meeting with the NRC staff, on October 16, 2002, to discuss the upcoming Waterford Steam Electric Station, Unit 3 (Waterford 3) extended power uprate. The meeting was useful to Entergy in better understanding areas of NRC staff and Advisory Committee on Reactor Safeguards (ACRS) concerns in the review of previously submitted industry extended power uprate requests. One recommendation from the meeting was for Entergy to submit a list of expected methodology changes needed in support of the Waterford 3 extended power uprate. This list, along with a brief discussion of the use of each methodology, is provided as an attachment to this letter.

Since the original licensing of Waterford 3, the NRC staff has developed and approved many new analytical methodologies. A number of these new methodologies will be used to perform the analysis of Waterford 3 at the new uprate power level, and thus, the Waterford 3 licensing basis will be updated to reflect these new methodologies. Specifically, Waterford 3 intends to adopt the 1999 Evaluation Model for Large Break Loss-of-Coolant Accident (LOCA) and the Combustion Engineering Nuclear Transient Simulation (CENTS) methodology for non-LOCA transient analyses. The new methodologies that will be used have previously been reviewed and approved by the NRC staff. The new methodologies that will be used in conjunction with the Waterford 3 extended power uprate are discussed in the Attachment, along with references regarding staff approval of each methodology.

This letter discusses methodologies expected to be used in support of the Waterford 3 extended power uprate license amendment request and does not contain any commitments.

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If you have any questions or require additional information, please contact D. Bryan Miller at 504-739-6692.

Sincerely,

K.J. Peters
Director, Nuclear Safety Assurance

KJP/DBM/cbh

Attachment: Methodology Changes

cc: E.W. Merschoff, NRC Region IV
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Attachment

W3F1-2002-0106

Methodology Changes

Methodology Changes

This attachment discusses the methodology changes that will be used in seeking NRC approval for the Waterford Steam Electric Station, Unit 3 (Waterford 3) extended power uprate from 3441 MWt to 3716 MWt rated thermal power. These methods will be appropriately addressed in the extended power uprate License Amendment Request to be submitted in 2003. These methods have been previously reviewed and approved by the NRC staff for Combustion Engineering (CE)-designed plants. The transient and Loss-of-Coolant Accident (LOCA) safety analysis methodologies to be used at Waterford 3 have been used at other CE plants with comparable or bounding operating conditions.

I. Combustion Engineering Nuclear Transient Simulation (CENTS) Nuclear Steam Supply System (NSSS) Simulation Code

The non-LOCA transient analysis will be based upon the use of the NRC-approved revision of the CENTS code rather than the current Combustion Engineering System Excursion Code (CESEC) III code. The CENTS code was submitted for review to the NRC and an approved topical report issued (Reference I.1). Volume 3 of Reference I.1 presents comparisons of CENTS to both CESEC predictions and NSSS plant data for typical Combustion Engineering NSSS two loop designs. Volume 4 of Reference I.1 presented a similar study for Westinghouse NSSS. The relative response of the CENTS predictions to these benchmarks were the subject of NRC staff review prior to approval of the CENTS code.

The constraints listed in the Safety Evaluation Report (SER) were reviewed. Constraint number 5 in the SER prevented the use of the CENTS code for the analysis of the Control Element Assembly (CEA) Ejection event. While the power uprate analyses will not use the CENTS code for predictions of the core\fuel response to the CEA Ejection, the CENTS code will be used to examine the peak Reactor Coolant System (RCS) pressure consequences of the CEA Ejection event. This clarification of the use of CENTS has been reviewed and approved by the NRC staff in Reference I.2.

Reference I.2 documented NRC approval of the transition from CESEC to CENTS for the current operation of the Palo Verde units. The Palo Verde power uprate project was similarly done with the CENTS code. References I.3 and I.4 document similar NRC approval for the use of CENTS for Arkansas Nuclear One, Unit 2 (ANO-2). Note the rated thermal power for Palo Verde of 3990 MWt exceeds the 3716 MWt at which the uprated Waterford 3 plant will operate. The core power density for Waterford 3 will be 103.0 w/cc, comparable to the 102.9 w/cc power density that ANO-2 has operated with after implementing its 7.5% power uprate.

II. Large Break Loss-of-Coolant Accident (LBLOCA) Emergency Core Cooling System (ECCS) Performance

The Waterford 3 extended power uprate LBLOCA ECCS performance analysis will be performed with the 1999 Evaluation Model (EM) version of the Westinghouse LBLOCA evaluation model for CE designed Pressurized Water Reactors (PWRs.) The current Waterford 3 LBLOCA ECCS performance analysis uses the 1985 EM version of the Westinghouse LBLOCA evaluation model for CE PWRs.

The 1999 EM is described in Reference II.1. The 1999 EM is NRC-approved for use in licensing applications for CE designed PWRs (Reference II.2). The 1999 EM was previously approved for use for the ANO-2 extended power uprate LBLOCA ECCS performance analysis (Reference II.3).

III. Small Break Loss-of-Coolant Accident (SBLOCA) ECCS Performance

The Waterford 3 extended power uprate SBLOCA ECCS performance analysis will be performed with the S2M version of the Westinghouse SBLOCA evaluation model for CE designed PWRs. This is the same evaluation model that is used in the current Waterford 3 SBLOCA ECCS performance analysis. The S2M is described in Reference III.1. It is NRC-approved for use in licensing applications for CE designed PWRs (Reference III.2).

The Waterford 3 extended power uprate SBLOCA ECCS performance analysis will credit automatic operation of the Atmospheric Dump Valves (ADVs). The current Waterford 3 SBLOCA ECCS performance analysis credits operation of the Main Steam Safety Valves (MSSVs), which begin opening at a steam generator pressure that is approximately 50 psi greater than the pressure at which the ADVs automatically open. The Waterford 3 ADVs are safety related. ADVs have been credited in previous SBLOCA ECCS performance analyses at other plants (e.g., South Texas Project, Reference III.3).

IV. Post-LOCA Long Term Cooling (LTC)

The Waterford 3 extended power uprate post-LOCA LTC analysis will be performed with the LTC evaluation model described in CENPD-254 (Reference IV.1). This is the same LTC evaluation model that is used in the current Waterford 3 LTC analysis. CENPD-254 is NRC-approved for use in licensing applications for CE designed PWRs (Reference IV.2).

The boric acid precipitation portion of the extended power uprate LTC analysis will use a value for the mixing volume that is equal to the water volume from the top of the core support plate (i.e., the plate upon which the fuel assemblies rest) to the bottom of the hot leg nozzle that is inside the core baffle and, above the core baffle, that is inside the core barrel. The water volume in the lower plenum will not be included in the mixing volume. The value for the mixing volume used in the current Waterford 3 boric acid precipitation analysis includes the water volume in the lower plenum of the reactor vessel.

V. Statistical Convolution

Entergy intends to use the method of statistical convolution to predict the number of fuel pins which experience Departure from Nucleate Boiling (DNB.) This methodology is described in Reference V.1 and has been the basis of the Waterford 3 calculation of the determination of the number of fuel pins experiencing DNB for the Sheared Shaft\Seized Rotor event and Excess Load with Loss of Alternating Current (LOAC) event since the original licensing of the plant. Entergy intends on extending the use of this method to other events consistent with other CE plants.

The use of convolution has been approved for use at Calvert Cliffs for the fuel failure predictions for the Pre-trip Steam Line Break (SLB) event in 1995, Reference V.2. References V.3 and V.4 show NRC concurrence with the use of statistical convolution for the prediction of fuel failure for CEA Ejection and Seized Rotor\Sheared Shaft for System 80 and System 80+, respectively.

VI. Radiological Consequences

As part of the Waterford 3 extended power uprate fuel failure targets for the power uprate reload cores will be back-calculated from the corresponding acceptance criteria. In this way, the future cores will have a bounding amount of fuel failure as a limit they cannot exceed. The power uprate report will report this fuel failure percentage and the corresponding acceptance criteria. This approach was reviewed and approved by the NRC for use at St. Lucie in Reference VI.1

VII. Return to Power Steam Line Break Fuel Analyses

The uprate core will result in a more negative Moderator Temperature Coefficient (MTC) at end of cycle conditions. This is expected to somewhat adversely impact the analysis results of the Return-to-Power Steam Line Break (RTP SLB) event, in that fuel failure will now be calculated to occur for this event. The Macbeth correlation, Reference VII.1, is the current DNB Ratio (DNBR) correlation used for the RTP SLB. The extended power uprate work will continue to use Macbeth to determine DNBR related fuel failure for the RTP SLB. Application of the Macbeth correlation to the evaluation of the fuel performance takes place independently of the NSSS simulation code (i.e., CENTS or CESEC) used. The Macbeth correlation combines inputs from the global NSSS response (core average power, system pressure, etc) from the simulation code with detailed core power distribution information from the nuclear design codes in making the DNBR determination. The RTP SLB event was one of those specifically examined in the CENTS topical report. The range of the plant response values obtained with CENTS was very similar to that predicted by the CESEC code. Therefore the results of the RTP SLB will remain within the parameter range over which the Macbeth correlation has been quantified.

Previously, fuel failure for the RTP SLB has been reported and found acceptable by the NRC staff in References VII.2 and VII.3.

VIII. DDIFF Sub-compartment Analyses

The current steam generator sub-compartment pressurization analysis utilized the RELAP-3 computer code to calculate sub-compartment pressurization as a function of time for various postulated pipe breaks in the compartment. For extended power uprate, the DDIFF computer code will be used to calculate pressurization of the steam generator sub-compartment as a function of time for various postulated RCS tributary line breaks and a feedwater line break in the compartment. DDIFF was used to calculate steam generator subcompartment pressures for RCS tributary line breaks and a feedwater line break for the ANO-2 replacement steam generator project as described in Reference VIII.1. Approval of DDIFF is documented in the SER transmitted by NRC in Reference VIII.2.

IX. Containment Subcompartment – Leak-Before-Break (LBB)

The recalculation of RCS pipe break loads for power uprate conditions will be done taking credit for LBB per the NRC-approved CE topical report, Reference IX.1. The application of LBB means that the major components and supports of the RCS and attached piping will be re-evaluated for double ended branch line pipe breaks instead of the main loop pipe breaks currently in the design basis. LBB will not be used for evaluation of subcompartment floors and walls. The original design basis breaks currently in the Waterford 3 Final Safety Analysis Report will be used for re-evaluation of the design of the subcompartments under Power Uprate conditions.

X. Loss of Offsite Power

The transient analysis will assume a LOAC power in assessing the release paths for radioisotope release and the timing\extent to which engineered safety features respond to the events. Currently, only the Steam Generator Tube Rupture (SGTR) analysis assumes a 3 second delay between the turbine trip and the loss of non-safety related AC power for the purpose of demonstrating acceptable DNBR performance. This assumption will be retained for the SGTR re-analysis for extended power uprate. Additionally, note that the NRC has accepted the use of this delay (discussed in Reference X.1) for the SGTR event in Reference X.2 and for the Seized Rotor\Sheared Shaft event in Reference X.3. However, as stated above, Entergy only intends to retain this assumption for the Waterford 3 SGTR re-analysis.

XI. Summary

This letter discusses the methods planned to be used in obtaining NRC approval for Waterford 3 extended power uprate to a rated thermal power of 3716 MWt. The methods discussed have been previously reviewed and approved by the NRC for CE design plants. Transient and LOCA safety analysis methodology being used at Waterford 3 has been used at CE plants with comparable or bounding operating conditions.

XII. References

I.1 "Technical Manual for the CENTS Code," CENPD 282-P-A, February 1991.

I.2 Letter from L. R. Wharton (USNRC) to G. R. Overbeck (APS), "Palo Verde Nuclear Generating Station, Units 1, 2 and 3 – Issuance of Amendments Re: Various Administrative Controls (TAC Nos. MB1668, MB1669 and MB1670)," October 15, 2001.

I.3 "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 189 to Facility Operating License No. NPF-6, Entergy Operations, Inc. Arkansas Nuclear One, Unit No. 2, Docket No. 50-368."

I.4 "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 190 to Facility Operating License No. NPF-6, Entergy Operations, Inc. Arkansas Nuclear One, Unit No. 2, Docket No. 50-368."

II.1 CENPD-132, Supplement 4-P-A, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," March 2001.

II.2 Letter from S.A. Richards (NRC) to P.W. Richardson (Westinghouse), "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.

II.3 Letter from T.W. Alexion (NRC) to C.G. Anderson (Entergy), "Arkansas Nuclear One, Unit No. 2 – Issuance of Amendment Re: Increase in License Power Level (TAC No. MB0789)," April 24, 2002.

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

III.2 Letter form T.H. Essig (NRC) to I.C. Rickard (ABB CENP), "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.

III.3 Letter from T.W. Alexion (NRC) to W.T. Cottle (STP), "South Texas Project, Units 1 and 2 – Issuance of Amendments Re: Operator Action for Small-Break Loss-of-Coolant Accident (TAC Nos. MA2498 and MA2499)," December 14, 1999.

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

IV.2 Letter from R.L. Baer (NRC) to A.E. Scherer (CE), "Staff Evaluation of Topical Report CENPD-254-P," July 30, 1979.

V.1 "Loss of Flow, C-E Methods for Loss of Flow Analysis," CENPD-183-A

V.2 Letter from D.G. McDonald (NRC) to R.E. Denton (BG&E) "Approval To Use Convolution Technique In Main Steam Line Break Analysis – Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2," May 11, 1995.

V.3 "Safety Evaluation Report related to the final design of the Standard Nuclear Steam Supply Reference System CESSAR System 80, Docket No. STN-50-470" NRC, November 1981.

V.4 "Final Safety Evaluation Report Related to the Certification of the System 80+ Design," USNRC, August 1994.

VI.1 "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 105 to Facility Operating License No. NPF-18. Florida Power and Light Company et. al. St. Lucie Plant, Unit No. 2, Docket No. 50-389"

VII.1 "An Appraisal of Forced Convection Burnout Data," Proc. Inst. Mech. Eng., Vol. 180, Pt 3c, PP 37-50, 1965-1966.

VII.2 Letter from B.T. Moroney (NRC) to J.A. Stall (FP&L), "St. Lucie Plant, Unit No. 2 – Issuance of Amendment Regarding Revised Post-Trip Steam Line Break Analysis (TAC No. MB0616)," June 19, 2001.

VII.3 "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 104 to Facility Operating License No. DPR-53, Baltimore Gas and Electric Company, Calvert Cliffs Nuclear Power Plant Unit No. 1, Docket No. 50-317"

VIII.1 "Arkansas Nuclear One – Unit 2 Replacement Steam Generator Project, Replacement Steam Generator Report," May 2000

VIII.2 Letter from K. Kniel (NRC) to A.E. Scherer (CE), "Evaluation of Topical Report CENPD-141 Revision 1, January 1977"

IX.1 Combustion Engineering Owners Group Report CEN-367, "Leak-Before-Break Evaluation of Primary Coolant Loop Piping in Combustion Engineering Designed Nuclear Steam Supply Systems", November 1987

X.1 CE letter LD-82-040 dated March 31, 1982, to USNRC, "Turbine Trip Time Delay"

X.2 "Safety Evaluation Report Related to the Final Design of the Standard Nuclear Steam Supply Reference System CESSAR System 80, Docket No. STN-50-470" NUREG-0852, Supplement No. 1, USNRC, March 1983.

X.3 "Safety Evaluation Report Related to the Final Design of the Standard Nuclear Steam Supply Reference System CESSAR System 80, Docket No. STN-50-470" NUREG-0852, Supplement No. 2, USNRC, September 1983.