Entergy

Entergy Nuclear South
Entergy Operations, Inc.
17265 River Road
Killona, LA 70057
Tel 504 739 6440
Fax 504 739-6698
kpeters@entergy.com

Ken Peters
Director, Nuclear Safety Assurance
Waterford 3

W3F1-2004-0073

August 25, 2004

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

SUBJECT: Supplement to Amendment Request NPF-38-249,
Extended Power Uprate
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

REFERENCES:
1. Entergy Letter dated November 13, 2003, "License Amendment Request NPF-38-249 Extended Power Uprate"
2. NRC Letter dated June 21, 2004, "Waterford Steam Electric Station, Unit 3 (Waterford 3) – Request for Additional Information Related to Revision to Facility Operating License and Technical Specifications - Extended Power Uprate Request (TAC No. MC1355)"
3. Entergy Letter dated July 28, 2004, "Supplement to Amendment Request NPF-38-249 Extended Power Uprate"
4. Entergy Letter dated August 10, 2004, "Supplement to Amendment Request NPF-38-249 Extended Power Uprate"
5. Entergy Letter dated July 14, 2004, "Supplement to Amendment Request NPF-38-249 Extended Power Uprate"

Dear Sir or Madam:

By letter (Reference 1), Entergy Operations, Inc. (Entergy) proposed a change to the Waterford Steam Electric Station, Unit 3 (Waterford 3) Operating License and Technical Specifications to increase the unit's rated thermal power level from 3441 megawatts thermal (MWt) to 3716 MWt.

By letter (Reference 2), the Nuclear Regulatory Commission (NRC) staff requested additional information (RAI) related to reactor systems. By letters (Reference 3 and Reference 4) Entergy responded to 60 of the 61 questions and committed to provide the remaining response in a future supplement. Entergy's response to the last unanswered question is contained in Attachment 1 to this letter.

The Waterford 3 Extended Power Uprate (EPU) Power Uprate Report (PUR) was submitted as Attachment 5 to Reference 1. The control element assembly (CEA) ejection analysis

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presented in Section 2.13.4.3.2 of the PUR included the results of a peak reactor coolant system (RCS) pressure case. The peak pressure case has been reanalyzed to conservatively prevent actuation of the pressurizer sprays and to model the occurrence of a turbine trip following reactor trip. The revised analysis indicates that peak calculated RCS pressure increased from 2519 psia previously reported in Reference 1 to 2613 psia, a value less than the acceptance criterion of 2750 psia (i.e., 110% of RCS design pressure). Revised PUR pages are provided in Attachment 2 and supersede the corresponding pages previously provided in Reference 1.

Following conference calls with members of the NRC staff, Entergy is providing additional information in support of the EPU.

- Attachment 3 contains additional information regarding the EPU containment analysis.
- Attachment 4 contains additional information regarding the development of the EPU steam generator pressure – low setpoint and its allowable value. Note that an arithmetic error was identified that results in a steam generator pressure – low allowable value different from that proposed in Reference 1. The error has been entered into Entergy's 10 CFR 50 Appendix B Corrective Action Program and has been corrected in the attached calculations. Revised technical specification mark-ups for technical specification pages 2-3, 3/4 3-19, and 3/4 3-20 will be provided in a future supplement to replace those previously provided.
- Attachment 5 contains additional information regarding the EPU spent fuel cooling analysis.

The no significant hazards consideration included in Reference 5 is not affected by any information contained in this letter. The submittal includes one new commitment as summarized in Attachment 6.

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 25, 2004.

Sincerely,

KJP/DBM/cbh

Attachments:
1. Response to Request for Additional Information
2. Revised Control Element Assembly Ejection Peak Pressure Analysis Results
3. Additional Information Related to EPU Containment Analysis
4. Additional Information Regarding EPU Steam Generator Pressure – Low Setpoint Development
5. Additional Information Regarding EPU Spent Fuel Pool Cooling Analysis
6. List of Regulatory Commitments

cc: Dr. Bruce S. Mallett
U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011

NRC Senior Resident Inspector
Waterford 3
P.O. Box 822
Killona, LA 70057

U.S. Nuclear Regulatory Commission
Attn: Mr. Nageswaran Kalyanam MS O-07D1
Washington, DC 20555-0001

Wise, Carter, Child & Caraway
Attn: J. Smith
P.O. Box 651
Jackson, MS 39205

Winston & Strawn
Attn: N.S. Reynolds
1400 L Street, NW
Washington, DC 20005-3502

Louisiana Department of Environmental Quality
Office of Environmental Compliance
Surveillance Division
P. O. Box 4312
Baton Rouge, LA 70821-4312

American Nuclear Insurers
Attn: Library
Town Center Suite 300S
29th S. Main Street
West Hartford, CT 06107-2445

Attachment 1 To

W3F1-2004-0073

Response to Request for Additional Information

Response to Request for Additional Information

Question 20:

Please provide a tabulation to indicate that for each event, what specific acceptance criteria are satisfied, to demonstrate that the general acceptance criteria of the event's class are met?

Response 20:

Table RAI 20-1 contains the regulatory limits per the NUREG-0800 Standard Review Plan acceptance criteria to which the events were evaluated and analyzed for the power uprate project.

TABLE RAI 20-1

EPU Section	Event	Pressure Criteria	Barrier Integrity
2.13.1.1.1	Decrease in Feedwater Temperature [1]	Max Pressures ≤ 110% of Design	No Specified Acceptable Fuel Design Limit (SAFDL) Violation Doses ≤ 10% of 10CFR100
2.13.1.1.2	Increase in Feedwater Flow [1]	Max Pressures ≤ 110% of Design	No SAFDL Violation Doses ≤ 10% of 10CFR100
2.13.1.1.3	Increased Main Steam Flow	Max Pressures ≤ 110% of Design	No SAFDL Violation Doses ≤ 10% of 10CFR100
2.13.1.1.4	Inadvertent Opening of a Steam Generator ADV (IOADV)	Max Pressures ≤ 110% of Design	No SAFDL Violation Doses ≤ 10% of 10CFR100
2.13.1.2.1	Decrease in Feedwater Temperature with Single Active Failure (SAF) [1]	Max Pressures ≤ 110% of Design	SAFDL Violation Permitted Doses ≤ 10% of 10CFR100
2.13.1.2.2	Increase in Feedwater Flow with SAF [1]	Max Pressures ≤ 110% of Design	SAFDL Violation Permitted Doses ≤ 10% of 10CFR100
2.13.1.2.3	Increased Main Steam Flow with SAF	Max Pressures ≤ 110% of Design	SAFDL Violation Permitted Doses ≤ 10% of 10CFR100
2.13.1.2.4	IOADV with Loss of Offsite Power (LOOP)	Max Pressures ≤ 110% of Design	SAFDL Violation Permitted [2] Doses ≤ 10% of 10CFR100
2.13.1.3.1	Steam System Piping Failures Post-Trip Analysis	Max Pressures ≤ 110% of Design	SAFDL Violation Permitted Doses ≤ 10CFR100 [3]
2.13.1.3.2	Mode 3 and 4 All Rods In (ARI) Return to Power (RTP) Steam Line Break (SLB)	Max Pressures ≤ 110% of Design	SAFDL Violation Permitted Doses ≤ 10CFR100
2.13.1.3.3.	Steam System Piping Failures Pre-Trip Power Excursion	Max Pressures ≤ 110% of Design	SAFDL Violation Permitted Doses ≤ 10CFR100 [3]
2.13.2.1.1	Loss of External Load [1]	Max Pressures ≤ 110% of Design	No SAFDL Violation Doses ≤ 10% of 10CFR100
2.13.2.1.2	Turbine Trip [1]	Max Pressures ≤ 110% of Design	No SAFDL Violation Doses ≤ 10% of 10CFR100
2.13.2.1.3	Loss of Condenser Vacuum	Max Pressures ≤ 110% of Design	No SAFDL Violation Doses ≤ 10% of 10CFR100

TABLE RAI 20-1

EPU Section	Event	Pressure Criteria	Barrier Integrity
2.13.2.1.4	Loss of Normal Alternating Current (AC) Power [1]	Max Pressures ≤ 110% of Design	No SAFDL Violation Doses ≤ 10% of 10CFR100
2.13.2.1.5	Steam Pressure Regulator Failure [1]	Max Pressures ≤ 110% of Design	No SAFDL Violation Doses ≤ 10% of 10CFR100
2.13.2.2.1	Loss of External Load with SAF [1]	Max Pressures ≤ 110% of Design	SAFDL Violation Permitted [2] Doses ≤ 10% of 10CFR100
2.13.2.2.2	Turbine Trip with SAF [1]	Max Pressures ≤ 110% of Design	SAFDL Violation Permitted [2] Doses ≤ 10% of 10CFR100
2.13.2.2.3	Loss of Condenser Vacuum with SAF	Max Pressures ≤ 110% of Design	SAFDL Violation Permitted [2] Doses ≤ 10% of 10CFR100
2.13.2.2.4	Loss of Normal AC Power with SAF [1]	Max Pressures ≤ 110% of Design	SAFDL Violation Permitted Doses ≤ 10% of 10CFR100
2.13.2.2.5	Loss of Normal Feedwater Flow	Max Pressures ≤ 110% of Design	No SAFDL Violation Doses ≤ 10% of 10CFR100
2.13.2.3.1	Feedwater System Pipe Breaks – Large Breaks	Max Pressures ≤ 120% of Design (Max Pressure ≤ 110% of Design for small FWLB with offsite power available)	SAFDL Violation Permitted [2] GIS Doses ≤ 10% of 10CFR100 PIS Doses ≤ 10% of 10CFR100 [5]
2.13.2.3.2	Loss of Normal Feedwater Flow with SAF	Max Pressures ≤ 110% of Design	SAFDL Violation Permitted [2] Doses ≤ 10% of 10CFR100
2.13.3.1.1	Partial Loss of Forced Reactor Coolant Flow [1]	Max Pressures ≤ 110% of Design	No SAFDL Violation Doses ≤ 10% of 10CFR100
2.13.3.2.1	Total Loss of Forced Reactor Coolant Flow	Max Pressures ≤ 110% of Design	No SAFDL Violation Doses ≤ 10% of 10CFR100
2.13.3.2.2	Partial Loss of Forced Reactor Coolant Flow with SAF [1]	Max Pressures ≤ 110% of Design	SAFDL Violation Permitted Doses ≤ 10% of 10CFR100
2.13.3.3.1	Single Reactor Coolant Pump (RCP) Shaft Seizure/Sheared Shaft	Max Pressures ≤ 110% of Design	SAFDL Violation Permitted Doses ≤ 10% of 10CFR100

TABLE RAI 20-1

EPU Section	Event	Pressure Criteria	Barrier Integrity
2.13.4.1.1	Uncontrolled Control Element Assembly (CEA) Withdrawal from Subcritical	Max Pressures ≤ 110% of Design	No SAFDL Violation Doses ≤ 10% of 10CFR100
2.13.4.1.2	Uncontrolled CEA Withdrawal from Low Power	Max Pressures ≤ 110% of Design	No SAFDL Violation Doses ≤ 10% of 10CFR100
2.13.4.1.3	Uncontrolled CEA Withdrawal at Power	Max Pressures ≤ 110% of Design	No SAFDL Violation Doses ≤ 10% of 10CFR100
2.13.4.1.4	CEA Misoperation	Max Pressures ≤ 110% of Design	No SAFDL Violation Doses ≤ 10% of 10CFR100
2.13.4.1.5	Inadvertent Boron Dilution	Max Pressures ≤ 110% of Design	No SAFDL Violation Doses ≤ 10% of 10CFR100 Time Between Alarm and Loss of Shutdown Margin: ≥ 30 Minutes, Mode 6 ≥ 15 Minutes, other Modes
2.13.4.1.6	Startup of an Inactive Reactor Coolant Pump	Max Pressures ≤ 110% of Design	No SAFDL Violation Shutdown Margin
2.13.4.1.7	CEA Withdrawal Modes 3, 4 and 5 ARI	Max Pressures ≤ 110% of Design	No SAFDL Violation Doses ≤ 10% of 10CFR100
2.13.4.3.1	Inadvertent Loading of a Fuel Assembly into an Improper Position	Max Pressures ≤ 110% of Design	SAFDL Violation Permitted [2] Doses ≤ 10% of 10CFR100
2.13.4.3.2	Control Element Assembly Ejection	Max Pressures ≤ 110% of Design	SAFDL Violation Permitted Doses ≤ 10CFR100
2.13.5.1.1	Chemical & Volume Control System (CVCS) Malfunction	Max Pressures ≤ 110% of Design	No SAFDL Violation Doses ≤ 10% of 10CFR100
2.13.5.1.2	Inadvertent Emergency Core Cooling System (ECCS)	Max Pressures ≤ 110% of Design	No SAFDL Violation Doses ≤ 10% of 10CFR100
2.13.5.2.1	CVCS Malfunction with SAF	Max Pressures ≤ 110% of Design	No SAFDL Violation Doses ≤ 10% of 10CFR100

TABLE RAI 20-1

EPU Section	Event	Pressure Criteria	Barrier Integrity
2.13.6.3.1	Small Primary Line Break Outside Containment	Max Pressures ≤ 110% of Design	SAFDL Violation Permitted [2] GIS Doses ≤ 10% of 10CFR100 PIS Doses ≤ 10CFR100 [4]
2.13.6.3.2	Steam Generator Tube Rupture	Max Pressures ≤ 110% of Design	SAFDL Violation Permitted [2] GIS Doses ≤ 10% of 10CFR100 PIS Doses ≤ 10CFR100
2.13.6.4	Inadvertent Opening of a Pressurizer Safety Valve [1]	Captured in the Small Break LOCA Spectrum 10CFR50.46	
2.13.9.1.1	Asymmetric Steam Generator	Max Pressures ≤ 110% of Design	No SAFDL Violation Doses ≤ 10% of 10CFR100

Footnotes:

[1] As documented in Power Uprate Report (PUR) Table 2.13.0-1, this event is bounded by a different Final Safety Analysis Report (FSAR) event.

[2] Waterford 3 PUR analyses meet the radiological acceptance criteria by demonstrating no fuel failure for these events.

[3] The radiological consequences for the Return to Power Main Steam Line Break (MSLB) and Pre Trip Power Excursion MSLB are added together and compared to the acceptance limits of 10CFR100.

[4] Applicability of 10CFR100 criteria to Letdown Line Break Doses with pre-existing Iodine spike (PIS) was approved via January 8, 2003, letter from NRC to Waterford 3. (TAC No. MB3231). NUREG-0800 does not address the PIS case for this event.

[5] NUREG-0800 Section 15.2.8 acceptance criteria for Feedwater Line Break (FWLB) are a small fraction of 10CFR100 based on an accident generated iodine spike (GIS). The case of a Pre-existing iodine spike is not addressed. Waterford 3, via PUR Section 2.13.2.3.1.6, applied the small fraction acceptance criteria for the case of a Pre-existing Iodine spike.

Attachment 2 To

W3F1-2004-0073

Revised Control Element Assembly Ejection Peak Pressure Analysis Results

Revised Control Element Assembly Ejection Peak Pressure Analysis Results

None of the discussion included in Section 2.13.4.3.2 of the Power Uprate Report (PUR) (Reference 1, Attachment 5) or its subsections (Sections 2.13.4.3.2.1 through 2.13.4.3.2.6) are impacted by the revised peak Reactor Coolant System (RCS) pressure case. The discussion included in these sections does not directly report the Control Element Assembly (CEA) Ejection peak RCS pressure case results. Details of the peak RCS pressure case are only discussed in the tables and figures included with Section 2.13.4.3.2 of the PUR. Hence, only the PUR Section 2.13.4.3.2 tables and figures that report peak pressure case results are impacted. PUR tables and figures reporting "full power" CEA Ejection case results remain unchanged. Moreover, the initial assumptions for the CEA Ejection peak pressure case reported in PUR Table 2.13.4.3.2-3 were not changed for the new case performed. Hence this PUR table remains unchanged.

Changes to the PUR tables and figures are summarized below.

- PUR Tables 2.13.4.3.2-1, 2.13.4.3.2-2 and 2.13.4.3.2-3 are UNCHANGED.
- The information in PUR Table 2.13.4.3.2-4, the RCS Peak Pressure Sequence of Events, is replaced.
- PUR Table 2.13.4.3.2-5 is UNCHANGED.
- PUR Figures 2.13.4.3.2-1 through 2.13.4.3.2-6 are UNCHANGED.
- The core power vs. time response in PUR Figure 2.13.4.3.2-7 is replaced.
- The core heat flux vs. time response in PUR Figure 2.13.4.3.2-8 is replaced.
- The core coolant temperatures vs. time response in PUR Figure 2.13.4.3.2-9 is replaced.
- The RCS pressure vs. time response in PUR Figure 2.13.4.3.2-10 is replaced.
- The SG pressure vs. time response in PUR Figure 2.13.4.3.2-11 is replaced.
- The reactivity components vs. time response in PUR Figure 2.13.4.3.2-12 is replaced.

Table 2.13.4.3.2-4
CEA Ejection Peak RCS Pressure Sequence of Events

3716 MWt EPU Time (sec)	Event	3716 MWt EPU Setpoint / Value
0.00	Mechanical Failure of CEDM causes CEA to eject.	--
0.05	CEA fully ejected	--
0.07	CPC VOPT, % of Full Power	163
0.08	Maximum core power occurs, % of full power	187.0
0.699	Trip breakers open	--
1.299	CEA's begin to drop into core	--
2.9	Maximum RCS pressure, PSIA	2613*
4.8	CEA fully inserted, core power reduced to below 10% power	--

*2597 PSIA for BOC Cycle 1 HFP CEA Ejection.

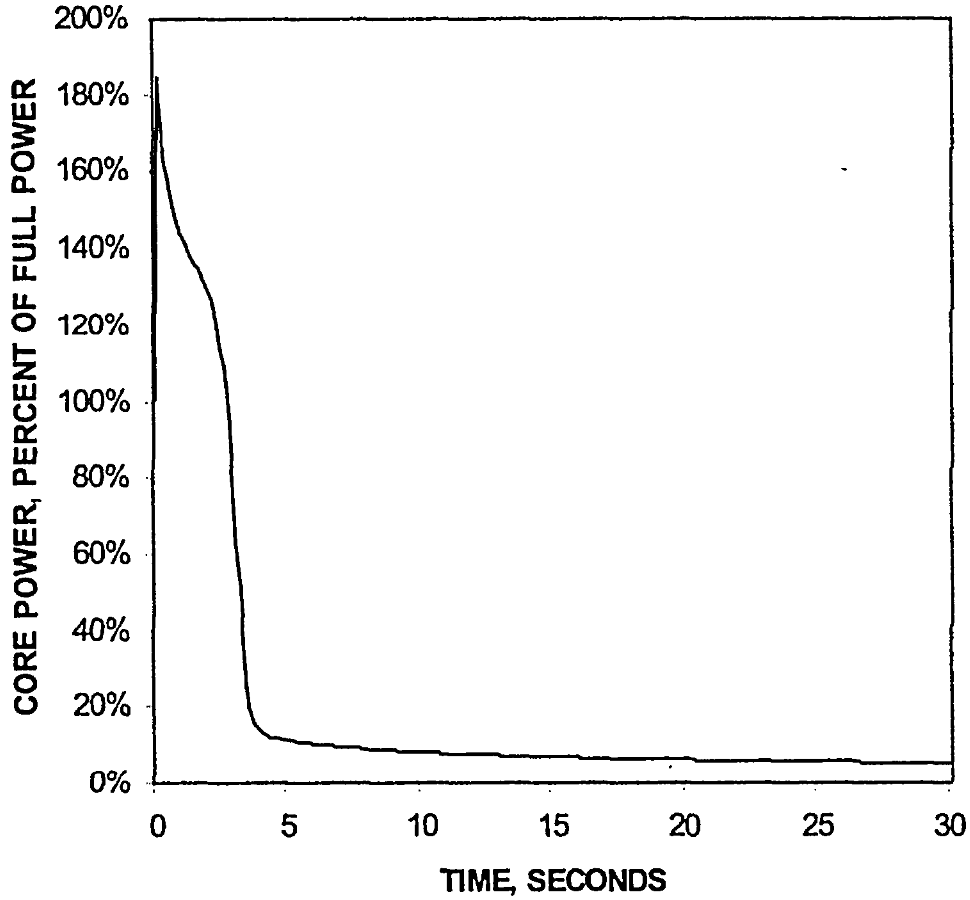

Figure 2.13.4.3.2-7
CEA Ejection Core Power vs. Time for Peak RCS Pressure

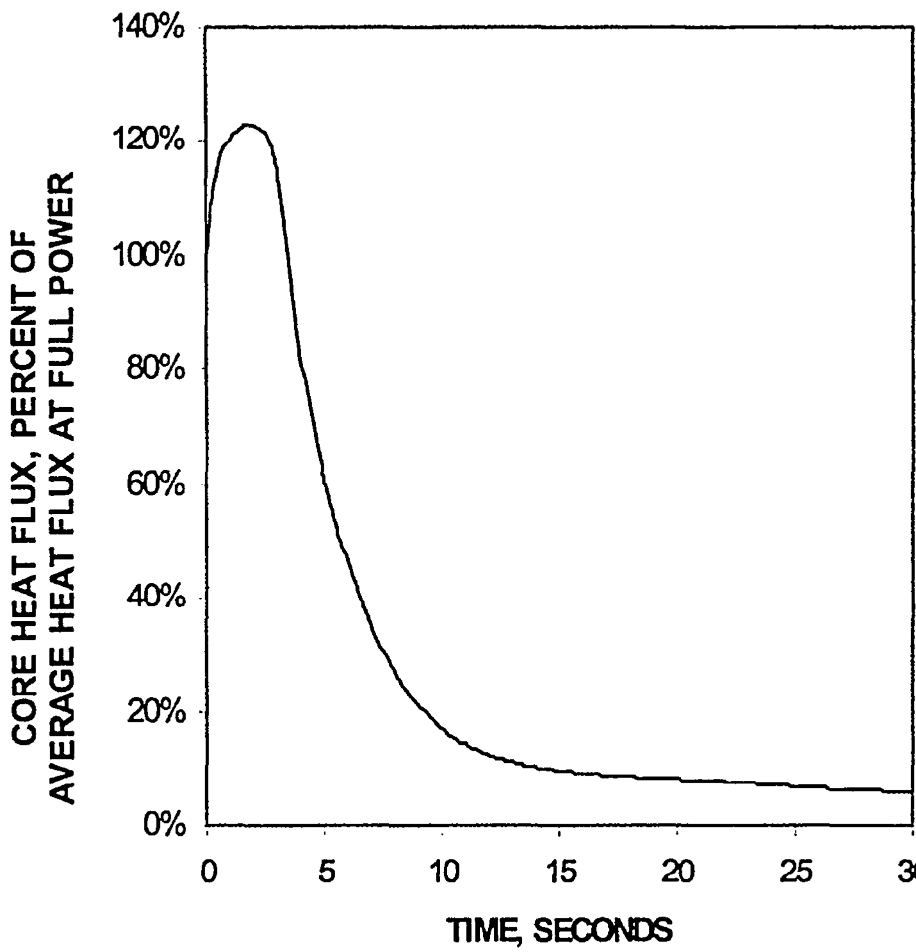

Figure 2.13.4.3.2-8
CEA Ejection Core Heat Flux vs. Time for Peak RCS Pressure

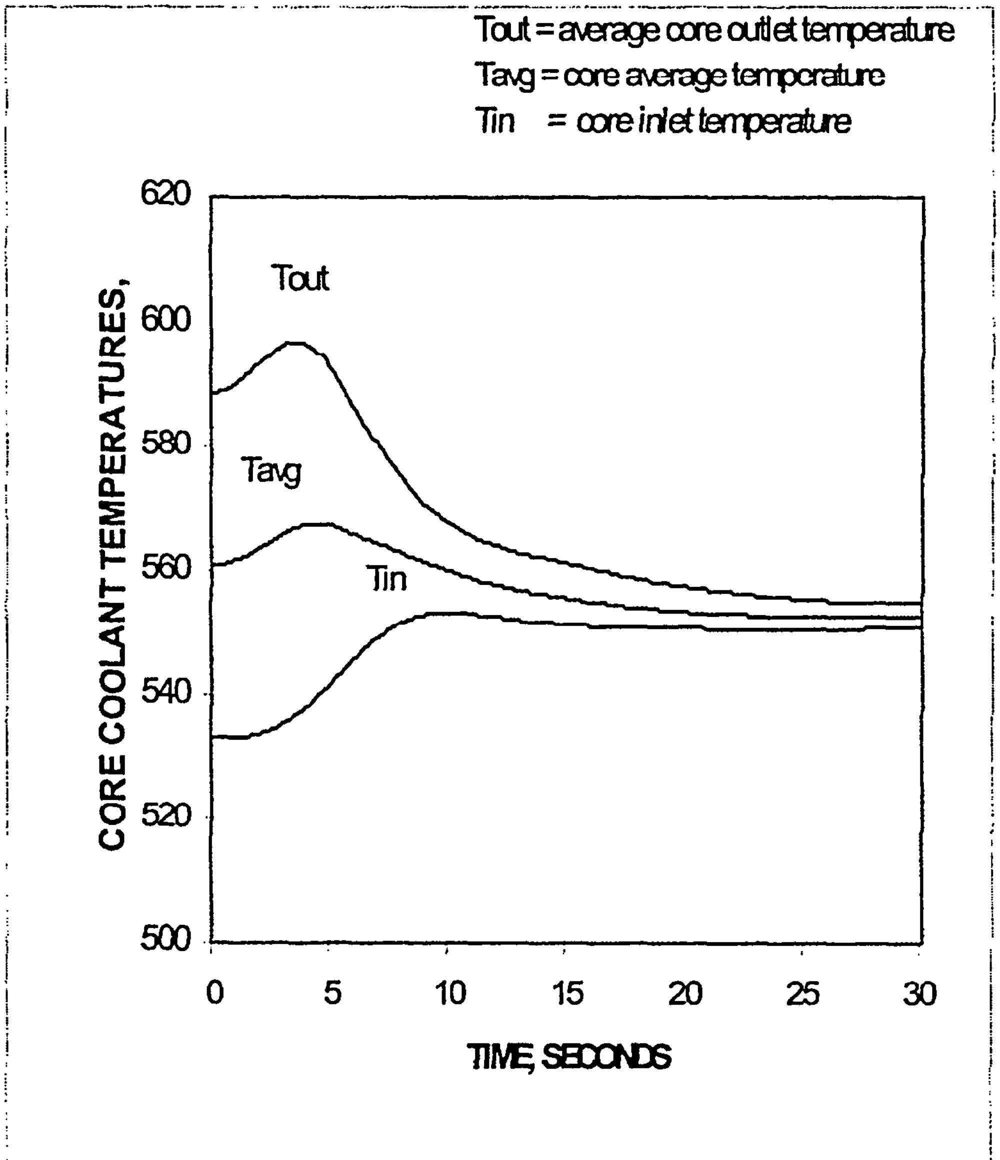

Figure 2.13.4.3.2-9
CEA Ejection Core Coolant Temperatures vs. Time for Peak RCS Pressure

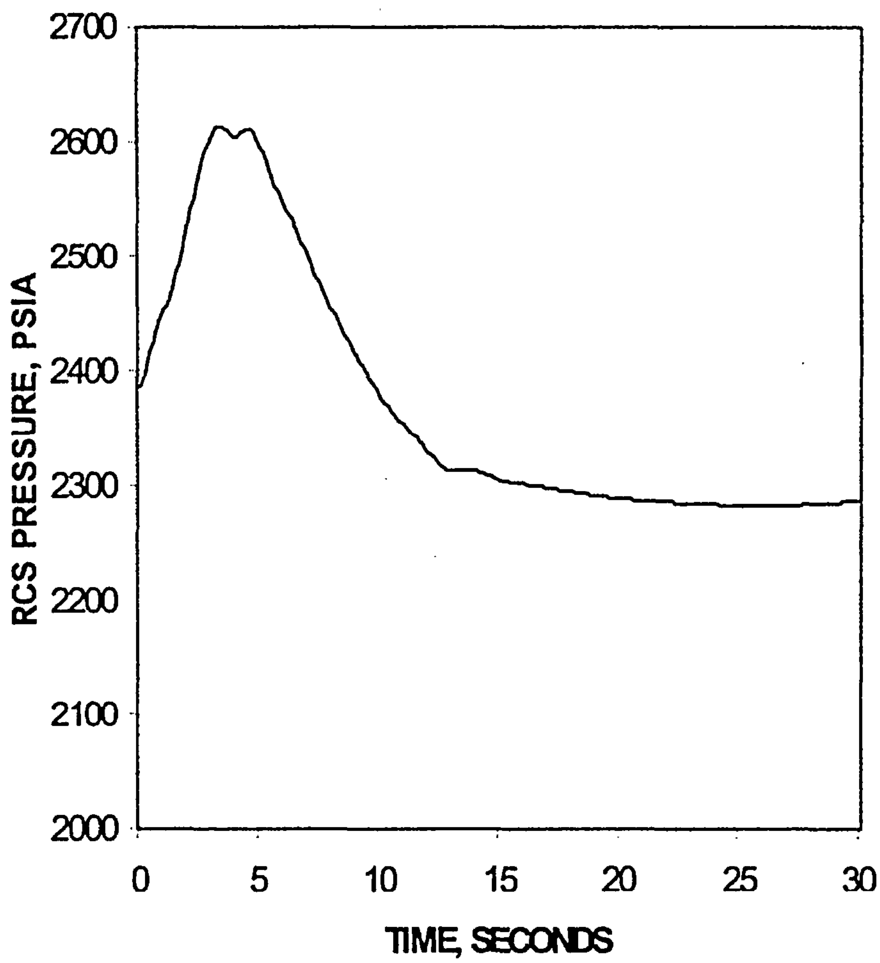

Figure 2.13.4.3.2-10
CEA Ejection Peak RCS Pressure vs. Time for Peak RCS Pressure

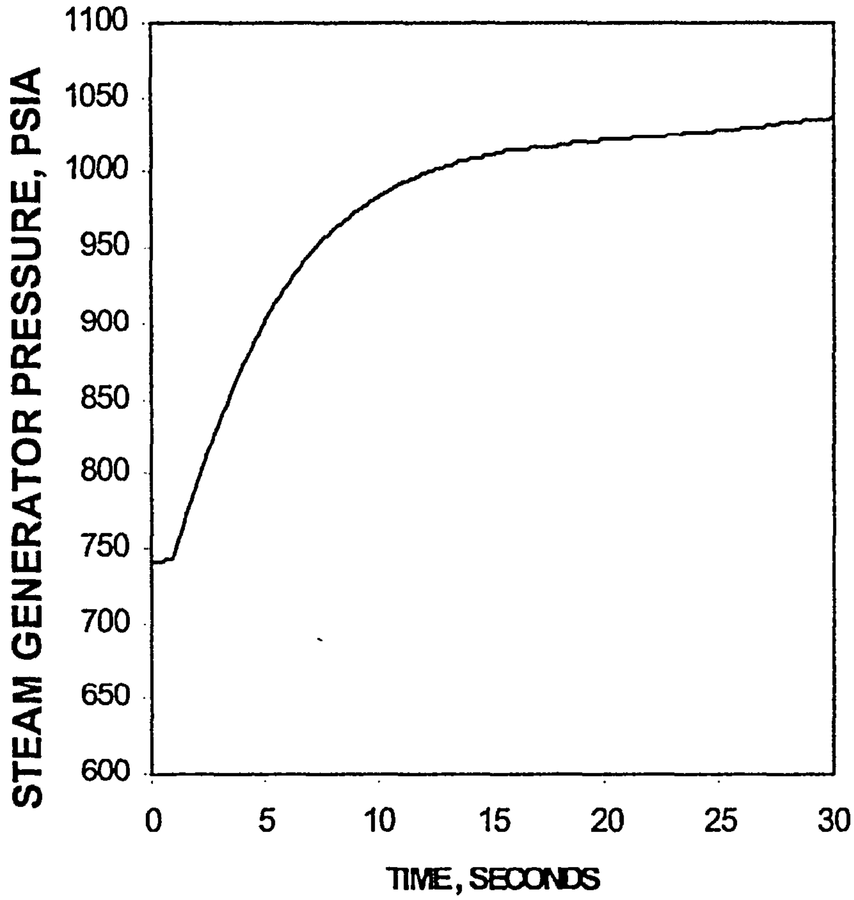

Figure 2.13.4.3.2-11
CEA Ejection SG Pressure vs. Time for Peak RCS Pressure

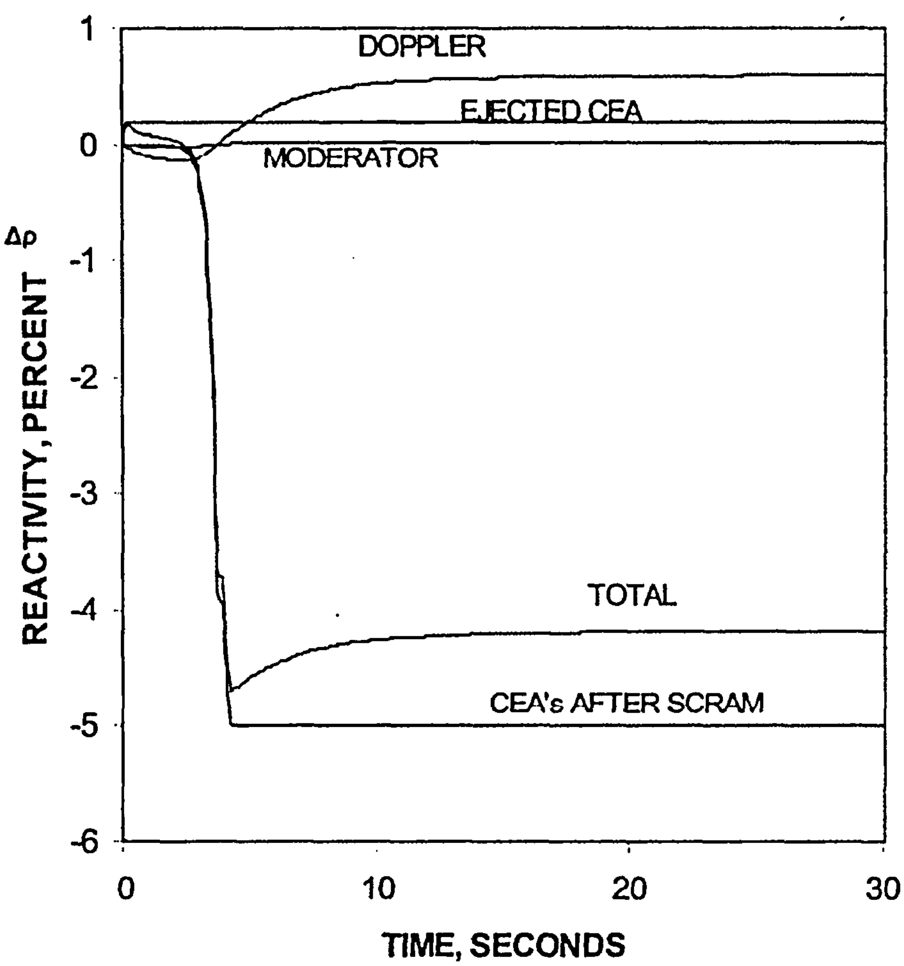

Figure 2.13.4.3.2-12
CEA Ejection Reactivity Components vs. Time for Peak RCS Pressure

Attachment 3 To

W3F1-2004-0073

Additional Information Related to EPU Containment Analysis

Additional Information Related to EPU Containment Analysis

Question 1:

An October 28, 1999 Waterford Unit 3 letter to the NRC provides a resolution to the penetration overpressurization issue of GL 96-06. A December 22, 1997 Waterford Unit 3 letter states that a containment atmosphere temperature of 260 F was assumed for the Waterford Unit 3 analyses. Please explain why this resolution is still valid for the power uprate. In particular, please explain why the pre-power uprate analyses remain valid when the peak temperature from the main steam line break is greater than 260 F.

Response 1:

The 260°F temperature reported in the December 22, 1997, letter was used to justify operability of the penetrations and continued operation during the period while penetrations susceptible to the overpressurization condition described in Generic Letter (GL) 96-06 could be reconfigured, modified, or drained to eliminate the potential for overpressurization. These changes, as committed to in response to GL 96-06, necessary to eliminate the potential for penetration overpressurization due to a temperature excursion from a DBA inside containment have been completed. Extended Power Uprate (EPU) does not propose to operate the containment penetrations differently or modify containment environmental parameters such that a containment penetration would become susceptible to conditions described in GL 96-06. EPU does not invalidate the measures taken to address the potential containment penetration overpressurization concern.

Question 2:

The power uprate submittal proposes using EQ temperature envelopes as the criteria for containment temperature for the main steam line break and LOCA analyses. (i) Explain why the containment design temperature of 263 °F is not used for the criterion. (ii) Since both the LOCA and main steam line break temperatures are greater than 263 °F, how is the containment design temperature satisfied?

Response 2:

(i) The originally determined peak loss of coolant accident (LOCA) and main steam line break (MSLB) temperatures (i.e., 269 °F and 413 °F respectively) were used in establishing the EQ temperature envelope as discussed in Section 3.11.3.3.1 of the original safety evaluation report. Thus the original licensing basis established the maximum allowed temperatures which were adopted as the acceptance criteria for the LOCA and MSLB. The adoption of the limits (i.e., 269.3 °F for LOCA and 413.5 °F for MSLB) are discussed in the Bases for Technical Specification 3/4.6.1.5, "Air Temperature." As discussed below, the containment steel vessel does not exceed the 263°F containment design temperature during the LOCA and MSLB events.

(ii) A calculation was done for Waterford 3 in 1985 to determine the maximum steel containment vessel wall temperature post-LOCA and post-MSLB. The "Maximum Steel Vessel (Containment) Temperature Estimation" calculation shows:

- For a peak LOCA temperature of 269.1°F the maximum containment vessel steel temperature would be 257.4°F, and
- for a peak MSLB temperature of 413.5°F the maximum containment vessel steel temperature would be 233.5°F.

The calculation notes: "The duration of the MSLB is very much smaller compared to LOCA. Therefore, LOCA conditions determine the maximum steel vessel temperature ..."

Therefore, since the EPU peak post-LOCA temperature (254.4°F) and the peak post-MSLB temperature (394.7°F) are less than those assumed in the maximum containment vessel steel temperature calculation discussed above, the maximum containment vessel steel temperature will remain below the "design temperature" of 263°F.