Entergy

Entergy Nuclear South
Entergy Operations, Inc.
17265 River Road
Killona, LA 70066
Tel 504 739 6440
Fax 504 739 6698
kpeters@entergy.com

Ken Peters
Director, Nuclear Safety Assurance
Waterford 3

W3F1-2004-0068

August 10, 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 14, 2004, "Supplement to Amendment Request NPF-38-249 Extended Power Uprate"
4. Entergy Letter dated July 28, 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 letter (Reference 4) Entergy responded to 40 of the 61 questions and committed to provide the remaining responses in a future supplement. Entergy's responses to 20 of the remaining 21 unanswered questions are contained in Attachment 1 to this letter. The response to the final unanswered question (i.e., #20) will be submitted by August 25, 2004.

Section 2.13 of Attachment 5 in Reference 1 presented the results of non- Loss of Coolant Accident (LOCA) transient analyses performed in support of the Extended Power Uprate (EPU). Section 2.13.3.3.1, "Single Reactor Coolant Pump (RCP) Shaft Seizure/Sheared Shaft," reported a fuel failure limit of 8%. Reexamination of the radiological consequences analysis for this event indicates that the fuel failure limit can be increased to 15% to better reflect the EPU approach of reporting in the licensing documentation the fuel failure amount

A001

which corresponds to the regulatory dose limits for the event. The EPU report page containing markups indicating this change is provided in Attachment 2.

In Section 2.5.5.3 of Attachment 5 and in Attachment 8 of Reference 1, Entergy committed to complete the evaluation of higher post-EPU component cooling water temperatures prior to operating at EPU conditions. Due to the complexity of the evaluations and corrective actions involved, Entergy proposes to revise this commitment to complete the evaluation and corrective actions following the implementation of the EPU relying on compensatory measures to insure acceptable system and plant protection in the interim. Attachment 3 provides additional discussion on the revised commitment.

The no significant hazards consideration included in Reference 4 is not affected by any information contained in this letter. The submittal includes one new commitment and one revised commitment as summarized in Attachment 4.

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 10, 2004.

Sincerely,

KJP/DBM/cbh

Attachments:
1. Response to Request for Additional Information
2. Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft
3. Component Cooling Water Design Temperatures
4. 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

W3F1-2004-0068

Response to Request for Additional Information

Response to Request for Additional Information

Question 1:

Please provide a quantified evaluation of the time needed for plant cooldown to achieve cold shutdown conditions per RSB BTP 5-1 (natural circulation cooldown using only safety grade equipment), and for plant cooldown per the requirements of Appendix R to Title 10 of the Code of Federal Regulations (10 CFR), Part 50 (regarding fire protection) for Waterford 3 at extended power uprate (EPU) power level and the current power level.

Response 1:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 2:

Clarify the statement in Section 2.6.4.4 of your EPU submittal, "the current limiting conditions for operations for time based reduced flow rate are acceptable at EPU condition."

Response 2:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 3:

Section 2.6.4.3 states that the low temperature overpressure (LTOP) calculations have been revised. Please submit: (1) the mass and energy input transient assumptions and results, (2) the revised LTOP enable setpoint, and (3) the vent capacities for NRC staff review.

Response 3:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 4:

The small break loss-of-coolant accident (SBLOCA) methodology (S2M) which was used to perform the Waterford 3 analyses for the uprated power does not apply at the requested uprated power, since the sensitivity studies supporting the S2M methodology were performed at a lower power. The sensitivity studies to justify applicability of the S2M at a higher power are plant-specific and do not have generic applicability. Provide justification that the S2M applies to Waterford 3 at the requested uprated power. (Please see the NRC staff safety evaluation report for Palo Verde Nuclear Generating Station fuel transition)

Response 4:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 5:

Please confirm that the generically approved LOCA analysis methodologies used for the Waterford 3 uprate LOCA analyses continue to apply specifically to the Waterford 3 plant by: 1) showing that Waterford 3 operating at the uprated power is bounded by the assumptions used in analyses used to support the approval of the generic LOCA methodologies identified in the response to Question 4; and 2) providing a statement to confirm that Waterford 3 and its vendor continue to have ongoing processes which assure that LOCA analysis input values bound the as-operated plant values for those parameters. (The statement should be identical to the one in the question in order to avoid providing extraneous and/or irrelevant information which will not address the question.)

Response 5:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 6:

What are the calculated large break LOCA and SBLOCA results per 10 CFR 50.46 (b) for Waterford 3 at the uprated power for both the new fuel and the resident fuel? Include in the evaluation of the local oxidation consideration of pre-event and post accident inside clad and outside clad oxidation.

Response 6:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 7:

Verify that the recently discovered error in the S2M methodology has been fixed in the Waterford 3 LOCA model.

Response 7:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 8:

Discuss the design of the Waterford 3 emergency core cooling system (ECCS) switch over from the injection mode to the ECCS sump recirculation mode. What was the decay heat source assumed in the design of the ECCS switch over from the injection mode to the ECCS sump recirculation mode for the present power? Does this assumed heat source change for the uprated power? Is the timing of the switch over affected? Please explain.

Response 8:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 9:

The Waterford 3 SBLOCA analyses take credit for the operation of the steam generator atmospheric dump valves (ADVs). Show that the ADVs are fully safety grade for this use by identifying all ADV components and supporting systems needed to support the SBLOCA operation, and show that these components and supporting systems are safety grade. (e.g, if the ADVs rely on instrument air, show that the instrument air system is safety grade and has sufficient long term capacity to support repeated cycling of the valves). If the valves are only qualified to open and not re-seat, show how the RCS pressure will be controlled to both keep the core covered and avoid cold overpressure for all break sizes. If stopping of ECCS pumps is involved show that the pumps can be re-started if and when needed.

Response 9:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 10:

a. Your February 5, 2004, Slide 5 stated that RCS flow rate would increase from 44,522 pound mass per second (lbm/sec) to 45,808 lbm/sec. Please provide the basis for this change.

b. Table 2.6-2 of your submittal states the minimum rate is unchanged from 148.0 X 10^6 pound mass per hour (41,411 lbm/sec) but it also states that the core and reactor vessel differential pressures increase in each of the Table's three columns and the nominal film coefficient is also shown to increase for the EPU. Please clarify this information with respect to ensuring that the 41,411 lbm/sec is bounding, that the film coefficient information is correct, and with respect to the Item 1.a values.

c. Please also address the stated core flow rates with respect to the above.

(We note similar information is provided elsewhere in your submittal, such as in Table 2.12-1.)

Response 10:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 11:

There are a number of references where the number of ADVs credited for the licensing bases is increased to two. Please address how this meets applicable single failure criteria.

Response 11:

The SBLOCA analysis that is described in Section 2.12.4 of the Extended Power Uprate Report (Attachment 5 to Reference 11.1) credits two ADVs. The analysis used the failure of an emergency diesel generator (EDG) as the most limiting single failure. Failure of an EDG results in the failure of one train of safety injection pumps. The failure of one ADV is less limiting than the failure of an EDG since both trains of safety injection pumps would remain operable following the failure of an ADV. However, subsequent to the submittal of the extended power uprate license amendment request, the failure of a DC power bus was identified as a more limiting single failure for the Waterford 3 SBLOCA analysis (Reference 11.2). It is more limiting than either the failure of an EDG or an ADV since it results in the consequential failure of both an EDG and an ADV. The SBLOCA analysis for the extended power uprate has been reanalyzed assuming the failure of a DC power bus (and, consequently, the failure of an EDG and an ADV). The analysis was provided to the NRC in Reference 11.3.

The post-LOCA long term cooling analysis for the extended power uprate (Section 2.12.5 of Attachment 5 to Reference 11.1) analyzed two different failure scenarios, namely, failure of an EDG (i.e., two ADVs operable) and failure of an EDG and one ADV (i.e., one ADV operable). Both analyses demonstrated acceptable post-LOCA long-term cooling. Only the results of the analysis that modeled two ADVs were described in Section 2.12.5 of the Extended Power Uprate Report.

One of the purposes of the post-LOCA long term cooling analysis is to demonstrate the ability to cooldown to where shutdown cooling can be initiated for those LOCAs that are small enough for the RCS to refill. The design temperature for the Shutdown Cooling System is 400°F, as documented in Final Safety Analysis Report (FSAR) Section 9.3.6.3.3. The analyses for the two different single failure scenarios used different values for the temperature to which the RCS was cooled down to prior to initiating shutdown cooling. The 1 ADV analysis cooled the RCS down to a hot leg temperature of 370°F, where 370°F conservatively bounds (i.e., is less than) the maximum indicated RCS temperature for initiating shutdown cooling (i.e., the shutdown cooling system design temperature of 400°F minus the RCS temperature measurement uncertainty). The 2 ADV analysis cooled the RCS down to a hot leg temperature of 350°F, where 350°F is the hot leg temperature for initiating shutdown cooling in the Waterford 3 LOCA emergency operating procedure. Both analyses initiated the steam generator/RCS cooldown at 1 hour post-LOCA and used a maximum steam generator/RCS cooldown rate of 40°F/hr.

Because of the larger area for steam relief, the 2 ADV analysis cooled the RCS down to 350°F faster than the 1 ADV analysis cooled the RCS down to 370°F despite the added stored energy associated with the additional 20°F of cooldown. The smaller amount of integrated decay heat removed during the shorter cooldown time more than offset the added stored energy associated with the additional 20°F of cooldown. Consequently, the 2 ADV analysis required less emergency feedwater to cooldown the RCS to 350°F than the 1 ADV analysis required to cooldown the RCS to 370°F. Both analyses modeled the steam generators as heat

sinks until 14 hours post-LOCA. At 14 hours, the 2 ADV analysis consumed more emergency feedwater since the integrated decay heat is the same at 14 hours for both analyses and the 2 ADV cooldown removed more stored energy. The emergency feedwater flow requirement identified in Table 2.12-12 of the Extended Power Uprate Report is the larger amount required by the 2 ADV analysis out to 14 hours post-LOCA. Note that the Waterford 3 Emergency Response Organization would be available to provide any assistance that might be required to ensure adequate heat removal, either through use of feedwater or via initiation of Shutdown Cooling.

Both the 1 ADV and 2 ADV analyses produced "long-term cooling plans" that demonstrated acceptable decay heat removal in the long-term for any break size. Figures 2.12-62 through 2.12-65 of Attachment 5 to Reference 11.1 present the long-term cooling plan and detailed results for the 2 ADV analysis. The results of the 1 ADV analysis are similar.

A similar analysis evolution occurred for the RSB 5-1 Natural Circulation Cooldown as with the SBLOCA, as described in the opening paragraph of this response. The analysis originally considered the two single failure scenarios of: 1.) failure of an ADV and 2.) failure of an EDG. When the failure of the DC power bus was identified as a potentially worse credible failure condition, the analysis was re-evaluated to further consider the consequences of the loss of control room control of one ADV in addition to the loss of one EDG. A description of the re-evaluated analysis was provided in the response to the NRC request for Additional Information, Question #1, as provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

The failure of an ADV to open is not necessarily postulated for cases where either ADV failure may not be the worst case single failure, may not be part of the worst case single failure, or for cases where it is not required to postulate single failures.

References

11.1 W3F1-2003-0074, J.E. Venable (EOI) to Document Control Desk (NRC), License Amendment Request NPF-38-249, Extended Power Uprate, Waterford Steam Electric Station, Unit 3, Docket No. 50-382, License No. NPF-38," November 13, 2003.

11.2 Event No. 40632, NRC Daily Events Report for April 1, 2004, "New Worst Case Single Failure May Exceed 10 CFR 50.46 Acceptance Criterion for SBLOCA."

11.3 W3F1-2004-0052, J.E. Venable (EOI) to Document Control Desk (NRC), "Supplement to Amendment Request NPF-38-249, Extended Power Uprate, Waterford Steam Electric Station, Unit 3, Docket No. 50-382, License No. NPF-38," July 14, 2004.

Question 12:

What was the previously assumed numerical value of the volume in which boric acid accumulates for long term cooling analysis? What is the numerical value assumed for the EPU?

Response 12:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 13:

Please confirm that the Table 2.12-12 values for the boric acid makeup tanks, refueling water storage pool, and safety injection tanks are those used in the long term cooling analyses.

Response 13:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 14:

Please state the time by which the procedure(s) will reasonably ensure initiation of hot leg injection and provide a copy of the procedure(s). Provide the basis for the stated time.

Response 14:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 15:

In your evaluation of non-LOCA transient analyses, there are a few events for which the following conclusion has been drawn, "The analysis has been evaluated for EPU, the final safety analysis report (FSAR) results remain bounding, and a complete reanalysis was not required." Please provide a quantitative evaluation of these events to show that the consequences of these events at EPU conditions are bounded by the current analysis in FSAR.

Response 15:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 16:

CENPD-282-P-A, Technical Manual for the CENTS Code, is referenced in W3F1-2003-0074 (November 13, 2003). Since then, the NRC staff has reviewed and accepted WCAP-15996-P, Technical Description Manual for the CENTS Code (on December 1, 2003), which includes certain updates to the CENTS code. Does the version of the CENTS code that has been used in the non-LOCA analyses include any updates made since 1995? If so, then the updated CENTS technical manual should be cited.

Response 16:

None of the modifications to the CENPD-282-P-A CENTS code described in WCAP-15996-P were employed in the modeling of the transients for the Waterford 3 EPU submittal.

Question 17:

Why is NUREG-75/087 (reference 2.13-14) cited, and not NUREG-0800.

Response 17:

NUREG-75/087 is the version of the standard review plan which was in force during the original licensing of Waterford 3.

Question 18:

Where is Figure 2.13-1

Response 18:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 19:

Do the classes of moderate frequency incidents, infrequent incidents, and limiting faults correspond to condition II, III and IV events of ANSI N18.2? What are the acceptance criteria that are applied in the EPU analyses and evaluations for the classes of moderate frequency incidents, infrequent incidents, and limiting faults?

Response 19:

The classification of the events is consistent with the original licensing basis for Waterford 3. This classification is also consistent with that used for San Onofre 2 and 3.

The acceptance criteria applied to the analyses in this application are also consistent with the current licensing basis and are not firmly based on event frequency. The seized rotor event and the main steam line break event are both limiting faults events. The acceptance criteria are not the same however. The seized rotor must demonstrate doses within a small fraction of 10CFR100 (10% or 30 REM thyroid), while the main steam line break must demonstrate doses within 10CFR100 (100% or 300 REM thyroid).

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:

The response to this question will be provided by August 25, 2004.

Question 21:

For all events, in which a reactor trip is assumed to occur does the negative reactivity insertion account for the most reactive CEA being stuck in the fully withdrawn position?

Response21:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 22:

In all analyses of post-trip thermal margin, especially in the steam line break (SLB) cases, is the minimum departure from nucleate boiling ratio (DNBR) calculated in the region of the assumed stuck CEA? How is that done?

Response 22:

The thermal margin conditions of the fuel long after the time of reactor trip is only a concern for the return-to-power SLB event in which post trip core power and radial peaking increases could occur due to reductions in subcriticality. For other events, the minimum DNBR occurs during the trip sequences.

Return-to-Power SLB

The core power distribution associated with the return-to-power steam line break conditions is modeled using a full core model with the most reactive CEA stuck fully withdrawn from the core. Additionally, the asymmetric core inlet temperature distribution associated with the complete blowdown of one of the steam generators and isolation of the other steam generator following MSIS is imposed on the model.

The power peaking resulting from this core model is combined with the core average thermal hydraulic response obtained from a transient simulation of the event. The response of the NSSS in this simulation is again driven by core reactivity feedback behaviors modeling the core in the N-1 condition.

The power peaking resulting from the stuck CEA configuration and the other core thermal hydraulic conditions are examined for approach to DNB using the Macbeth DNBR correlation. The use of the Macbeth correlation is consistent with the current Waterford 3 licensing basis for the return-to-power steam line break event.

Unlike the CE-1 correlation, used for the other events in the power uprate report, Westinghouse does not have the statistical relationship between a specific DNBR value below the Specified Acceptable Fuel Design Limit (SAFDL) and the probability of actually being in DNB. Therefore, for the return-to-power steam line break event, all fuel pins that are predicted to violate the Macbeth DNBR SAFDL are assumed to experience DNB. These pins are, therefore, predicted to experience clad failure for the purposes of the radiological consequences calculation.

Other Events

Events which have a minimum DNBR during the period of reactor trip are not subject to the high radial peaking occurring due to the stuck CEA. For these events, the combination of thermal hydraulic parameters, including an initial radial peak, which corresponds to a minimum initial thermal margin condition, is determined. The monitoring system ensures that the assumed minimum initial margin condition is maintained during plant operations.

Should the event itself result in a change in radial peak (such as CEA drop or CEA ejection), the initial radial peak is increased by the change in radial peak due to the event. Additionally,

changes in radial peak due to changes in coolant conditions are included in the determination of the radial peak at the time of minimum DNBR.

With the exception of the CEA ejection, these other events use the flat ARO pin census to evaluate fuel failure should the minimum DNBR be below the SAFDL. For the CEA ejection, the pin census corresponds to that resulting from being at the power dependent insertion limit (PDIL), with the modeled ejected CEA.

Question 23:

In 2.13.0.2. Initial Conditions, it is noted that non-safety grade systems, that would act to mitigate a transient were not credited. Were any non-safety grade systems, that would act to aggravate a transient, credited?

Response 23:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 24:

How is the decay heat determined and applied in the applicable analyses? What is the standard used?

Response 24:

For the non-LOCA transient analysis in Section 2.13 of the power uprate report, the CENTS code uses point kinetics to model reactor power until the value calculated in this fashion becomes less than the decay heat standard. The code then transitions and uses the value contained in the modeled decay heat standard. The Waterford 3 EPU non-LOCA transient analyses use the 1979 ANS decay heat standard with the application of a 2σ uncertainty.

Question 25:

Are all accident analyses and evaluations, presented in the application, cycle independent? Are all the accident analyses and evaluations that bound certain events of this application also cycle independent?

Response 25:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 26:

For event analyses that bound event analyses of different categories (e.g., reactor coolant pump shaft seizure, a Limiting Fault event, bounds a partial loss of forced reactor coolant flow, an Infrequent event), please identify the specific results and criteria that are compared in order to reach the bounding conclusion.

Response 26:

In cases where one event indicates that it is bounded by another event, the criteria used are that the event identified as bounding is of the same general physical effect, the analysis identified as bounding has the more severe initiator and is analyzed to demonstrate the same acceptance criteria.

Examples of this are:

Section 2.13.2.2.1, Loss of Load with single active failure (SAF), an infrequent event, points to Section 2.13.2.1.3, Loss of Condenser Vacuum, a moderate frequency event as the event which is bounding. Both of these events involve a sudden reduction in the removal of heat from the reactor coolant system. The Loss of Condenser Vacuum has the more severe initiator in that both steam flow and feedwater flow are terminated by the initiator. Additionally, as an infrequent event is pointing at the more restrictive moderate frequency event, the application of the potentially more restrictive acceptance criteria to the less frequent event would be conservative.

Section 2.13.3.1.1, the partial loss of forced circulation, points to Section 2.13.3.2.1, the total loss of forced circulation as being the event which provides bounding results, not the sheared shaft as stated in the question. Both of these events involve rapid reductions in forced circulation and therefore are comparable physical effects. In this case, admittedly, a moderate frequency event is being compared to an event of lower frequency, and therefore potentially an event which would allow for more adverse results.

Both of these events have been analyzed to the same acceptance criteria (no SAFDL violation). Hence, as the infrequent event creates a more severe physical effect (the 4 pump coastdown rather than a 2 pump coastdown), and both events have been analyzed to the same acceptance criteria, it is acceptable for the moderate frequency event to refer to the infrequent event as being bounding.

Question 27:

Table 2.13.0-2 indicates that the lower limit of the pressurizer safety valve (PSV) setpoint is 2425 pounds per square absolute (psia). Table 2.13.0-3 indicates that the reactor protection system (RPS) analytical setpoint for high pressurizer pressure is 2422 psia. This RPS setpoint could be much higher considering instrument uncertainties. Please discuss the consequences of a potential lifting of the PSV prior to RPS actuation, which would prevent a reactor trip from occurring.

Response 27:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 28:

The steam generator tube rupture (SGTR) analysis which assumed that a loss-of-offsite power (LOOP) occurs three seconds following reactor trip is non-conservative for the radiological consequences. This assumption is not consistent with the current licensing basis at Waterford. Please provide the results of a SGTR analysis assuming a LOOP occurs at the events initiation.

Response 28:

A loss of AC power, concurrent with the initiation of the break would result in an "immediate" reactor trip on low reactor coolant pump speed. The sequences of the opening of the ADVs and main steam safety valves (MSSVs) would be moved from the 485 second time frame to the 2 or 3 second time frame. Concentrations of radionuclides in the steam generator would therefore be lower during the period of time in which the affected steam generator was actively steaming to the atmosphere.

As documented in Power Uprate Report (PUR) Section 2.13.0.1, the SGTR analysis assumes that the Loss of Offsite Power occurs 3 seconds following reactor trip. This assumption is consistent with assumptions made on other CE NSSS plants, such as Palo Verde. This was discussed in letter W3F1-2002-0106 to the NRC dated December 16, 2002. Note that the NRC has accepted the use of this delay, discussed in CE letter LD-82-040 to the NRC dated March 31, 1982, in the Safety Evaluation Report for the CE System 80 design, NUREG-0852 Supplements No. 1 (for the SGTR event) and No. 2 (for the sheared shaft\seized rotor event).

Question 29:

For a SGTR accident, the most limiting single failure is to assume a stuck open ADV on the failed steam generator after it is automatically open following the event. Please explain why this assumption is not reflected in the sequence of event provided for this event.

Response 29:

The existing licensing basis for Waterford 3, as documented in FSAR Chapter 15, does not account for a single failure of a stuck open ADV for SGTR. No changes are proposed for power uprate. The Waterford 3 SGTR analyses are consistent with the standard review plan, which calls for assessing the consequences of an SGTR with and without a Loss of Offsite Power to determine that the more severe case has been considered. During original plant licensing, the Waterford 3 response to NRC FSAR Question 211.34 discussed that loss of offsite power was considered to be the worst assumed single failure affecting releases. For Waterford 3, if offsite power is available, the condenser is available as a steaming path through the Steam Bypass System, alleviating the need to cool down using the ADVs. Thus, the most severe case for Waterford 3 involves the assumption of a Loss of Offsite Power.

Further, provision is included in Waterford 3 operating procedures for local manual operation of the ADVs. Additionally, the Waterford 3 design includes ADV block valves upstream of the ADVs themselves. These block valves are available to isolate a stuck open ADV. Thus, there is no need to assume a single failure which results a stuck open ADV.

Question 30:

Table 2.13.0-1 indicates that the pressurizer safety valve lift transient is categorized as a "limiting fault" and bounded by the SBLOCA. Standard Review Plan 15.6.1 categorizes this event as an event of moderate frequency with the acceptance criteria associated with an event with moderate frequency occurrence. Please provide the results of an analysis for this event at EPU conditions to demonstrate that these acceptance criteria are met. Based upon its frequency of occurrence during "more than 260 pressurizer safety valve years of operation", and the observation that it could only be caused by a passive mechanical failure, does operating experience support the classification of this event as a faulted condition? Provide a tabulation of the thermal design parameters and compare them to the values assumed in safety analyses to demonstrate that the safety analyses assumptions are conservative.

Response 30:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 31:

Consider the event where one or both pressurizer safety valves were to open during a moderate frequency event (e.g., loss of condenser vacuum), and then fail to reseat properly. If this failure rate were to be high enough, then the analysis of the moderate frequency event would have to account for the effects of an open pressurizer safety valve. What failure rate has been assumed for the proper reseating of pressurizer safety valves in the analyses of events.

Response 31:

The failure of a pressurizer safety valve to reseat is not one of the single failures currently considered for Waterford 3. The extended power uprate project therefore did not consider this failure. Neither plants of CE nor Westinghouse design consider a failure of a pressurizer safety valve to close as a single failure.

Question 32:

Table 2.13.0-1 indicates that the Increased Steam Flow event (2.13.1.1.3) is analyzed as a moderate frequency event. The acceptance criteria, inter alia, specify that the resulting radiological dose must be less than or equal to a small fraction of 10 CFR 100 limits. Please quantify "small fraction". How does this radiological dose limit compare with the requirements of paragraph 20.1 of 10 CFR 20?

Response 32:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 33:

Section 2.13.1.1.3.1 states that any one of the following events may cause an increase in steam flow:

a) Inadvertent opening of the turbine admission valves. (approximately a 11% increase of the full power turbine flow rate)
b) Failure in the Steam Bypass System that could result in an opening of one steam bypass valve. (approximately 12.3% of the full power turbine flow rate)
c) Inadvertent opening of an ADV or SG safety valve. Each dump valve can release approximately 5.3% of the full-power steam flow, and the safety valve can pass approximately 9.3% of full power steam flow.

Failure of a steam bypass valve is declared to be the most adverse event. How has this determination been made? Were the reactivity effects of asymmetric core cooling, caused by the opening of one SG safety valve, considered?

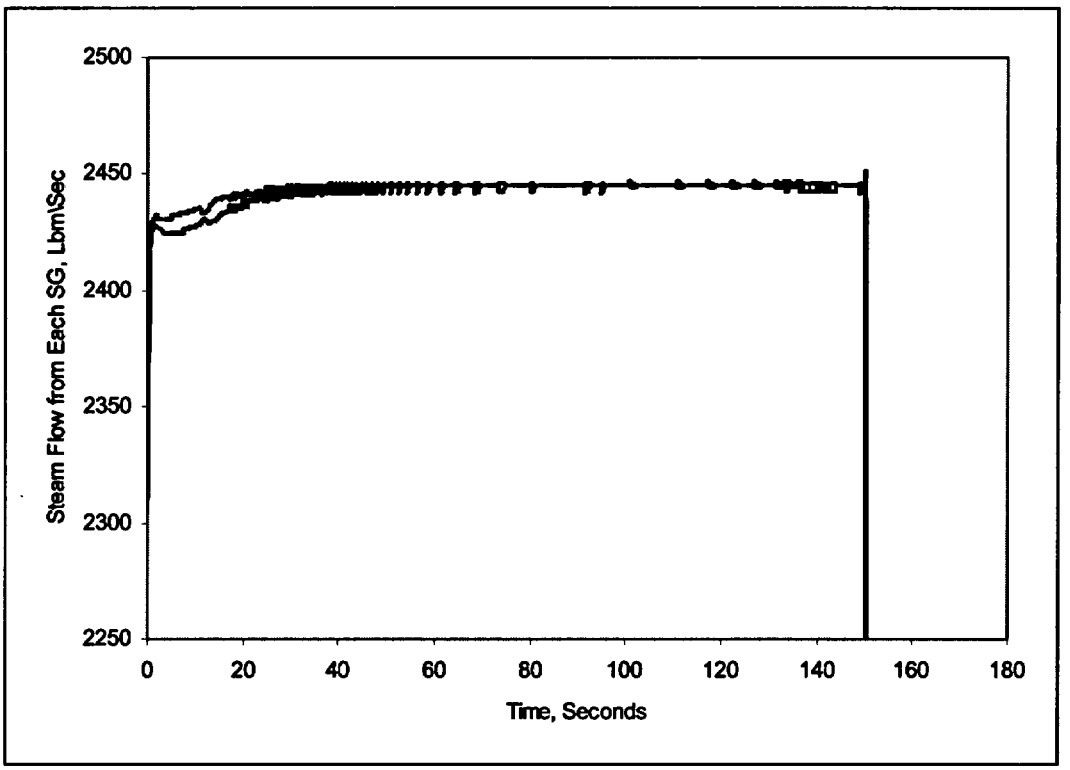
Response 33:

The failure of the steam bypass valve at full power conditions results in the greatest excess steam demand to the RCS. This results in the greatest cooldown of the core and highest positive reactivity insertion prior to trip.

The steam lines in the CENTS model contain the connection information of the main steam header. Figure RAI #33-1 is a high resolution plot of steam flow from the steam generators following the opening of an ADV on one steam line. It is seen that both steam generators share the additional steam demand, one via increased flow through the ADV and the other picking up more of the turbine demand. Therefore, the asymmetry caused by the opening of a valve on one steam line is very minor.

The exception to this is the asymmetric steam generator event, where one of the main steam isolation valves (MSIVs) is postulated to close. This closure of an MSIV terminates the communication through the main steam line header, as one steam generator picks up full steam demand and the other has steam flow terminated. For this event, the impact of the asymmetric core inlet conditions on the power distribution is explicitly modeled.

Figure RAI #33-1



Question 34:

Define RTP (rated thermal power) SLB (steam line break).

Response 34:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 35:

A major difference between the analysis results of the EPU SLB and the analysis results of the current SLB is that the DNBR SAFDL is violated (Tables 2.13.1.3.1-3 and 2.13.1.3.1-4). How is the extent of fuel pin failure (e.g. 2%) determined?

Response 35:

The system response to the return-to-power (RTP) steam line break (SLB) with loss of offsite power is determined. Due to the highly peaked power distribution associated with the core being in the N-1 configuration, the hottest fuel rods in the core are predicted to be in violation of the DNBR SAFDL. The transient analysis has also determined the radial peak, below which violation of the DNBR SAFDL will not occur.

A full core pin census is generated, with the core in the N-1 configuration, at the peak power conditions predicted to occur during the RTP SLB transient. It is verified that no more than 2% of the fuel pins will have a radial peaking factor greater than that which results in the DNBR SAFDL at the conditions of peak power during the RTP SLB.

The MacBeth DNBR correlation is used to predict the approach to DNB during the RTP SLB. All fuel pins which are predicted to have a MacBeth DNBR of less than the 1.30 SAFDL are assumed to fail. The method of assessing the thermal hydraulic behavior is consistent with that of the current licensing basis which is used to demonstrate that fuel failure does not occur.

Question 36:

Why is loss-of-normal feedwater flow (Section 2.13.2.2.5), considered to be an Infrequent Event, and not a Moderate Frequency Event?

Response 36:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 37:

Why is total loss-of-forced reactor coolant flow (Section 2.13.3.2.1), considered to be an Infrequent Event, and not a Moderate Frequency Event?

Response 37:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 38:

Why is Inadvertent loading of a fuel assembly into an improper position (Section 2.13.4.3.1), considered to be a Limiting Fault, and not an Infrequent Event?

Response 38:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 39:

Why is SGTR (Section 2.13.6.3.2), considered to be a Limiting Fault, and not an Infrequent or Moderate Frequency Event?

Response 39:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 40:

Why is small primary line break outside containment (Section 2.13.6.3.1), considered to be a Limiting Fault, and not an Infrequent Event?

Response 40:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 41: Section 2.6.1.3.1.1, Thermal Margin Analysis, indicates that the Modified Statistical Combination of Uncertainties (MSCU) methodology is applied in the analyses. The minimum DNBR SAFDL would be 1.26, as listed in the current Technical Specifications. However, the FSAR still refers to the prior DNBR SAFDL of 1.19 (e.g., in Section 15.3.1.1, Partial Loss of Reactor Coolant Flow). When Amendment No. 183 was issued, on March 29, 2002, the Safety Evaluation Report noted that the FSAR had not been updated, and advised the applicant to update the FSAR, in accordance to the requirements of 10 CFR 50.71. Please make the necessary updates to the FSAR. Please indicate the minimum DNBR SAFDL and the calculated minimum DNBR for all applicable accident analyses. Please verify that all events that are bounded by FSAR analyses, with respect to thermal margin, are comparable to FSAR analyses that applied the MSCU method.

Response 41:

As required by 10 CFR 50.71(e), the Waterford 3 FSAR will be updated to reflect changes as a result of the Extended Power Uprate following NRC approval and implementation.

For every fuel cycle since Cycle 1, and continuing through the extended power uprate, the analytical process ensures that fundamental acceptance criteria are met. The fundamental criterion is that the combination of the initial thermal margin and the action of the reactor protection system do not allow for violation of the SAFDL for certain events.

Additionally, plant configuration is examined to ensure that a judgment involving one event being bounded by another event remains valid for an upcoming cycle.

In the particular case questioned, due to the relative flow coastdown rates associated with the partial loss of forced reactor coolant flow, it was determined that the total loss of forced reactor coolant flow results in a more severe degradation of thermal margin. The reload process ensures that the total loss of reactor coolant flow does not violate the SAFDL. Thus, protection for the partial loss of forced reactor coolant flow is provided for without specific simulation of that event.

Therefore, all events are implicitly or explicitly verified against the DNBR SAFDL and underlying thermal hydraulic method (i.e. the 1.26 MSCU DNBR value) licensed for each particular cycle.

Question 42:

Provide a quantified evaluation of the impacts of the EPU to a core power level of 3716 megawatts thermal (MWt) on the ability of WSES 3 to cope with a Station Blackout (SBO) event. The evaluation should address the capacities of the condensate storage tank, turbine driven auxiliary feedwater pump, station batteries, and backup air supplies for air operated valves for decay heat removal and RCS cooldown during the time period of an SBO.

Response 42:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 43:

To support the results of the loss of normal feedwater transient, please provide the following information:

1. Discuss the need for a time delay of emergency feedwater (EFW) flow to steam generators while the plant is operated below 15% rated power.
2. The results of a loss of normal feedwater transient assuming that the EFW flow is delivered within one minute following the event to show the effect of overcooling at the beginning of the transient.
3. Discuss the provisions made in plant emergency operating procedures (EOPs) for controlling EFW at the beginning of the event to prevent excess cooldown during this event.
4. Discuss the phenomena involved that causes the RCS pressure to peak and then decrease prior to EFW flow being delivered to steam generators.

Response 43:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 44:

Please confirm that the event scenario of the SGTR thermal-hydraulic analysis is consistent with EOPs at Waterford 3.

Response 44:

As discussed in FSAR Section 15.6.3.2, the analyses for SGTR are based upon the plant specific procedures for responding to a SGTR. It had been recognized that the releases due to SGTR are significantly influenced by operator actions. The major operator actions in response to an SGTR are described in the FSAR. The timing of the operator actions assumed is based upon ANSI/ANS-58.8-1984, "American National Standard Response Design Criteria for Nuclear Safety Related Operator Actions." The first operator action is assumed to occur 7 minutes after reactor trip, with 2-minute intervals between discrete operator control actions during the 8-hour cooldown modeled. Note that the precise timing of the steps may not be that which would occur during the actual transient due to variations between actual plant response and the response assumed in the analysis. Further, because the event scenario is constructed to simulate early operator intervention which exacerbates the releases associated with this event, it is not the intent that plant emergency procedures attempt to mimic or bound the assumptions of this analysis. The timing associated with specific actions may change due to the power uprate initial conditions, but the underlying logic of the construction of assumptions has not been altered by EPU. The specific timing for power uprate conditions is provided in Table 2.13.6.3.2-2 of the PUR. Among the specific actions modeled are:

- Operators control EFW flow to maintain intact SG level between 68% and 71% wide range (WR)
- Plant cooldown by steaming through the unaffected SG
- Isolation of the affected SG
- Initiation of auxiliary spray to depressurize the RCS

Question 45:

Provide the results of a SGTR thermal-hydraulic analysis to demonstrate that the SG will not be overfilled by EFW flow during this event.

Response 45:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 46:

Please provide a tabulation of all computer codes and methodologies used in the re-analyses and indicate the staff approval status, any conditions and limitations on their use, and how the limitations are satisfied for application at Waterford 3.

Response 46:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 47.

Provide a tabulation of the thermal design parameters and compare them to the values assumed in safety analyses to demonstrate that the safety analyses assumptions are conservative.

Response 47:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 48:

Expand Table 2.13.0-2 to include all primary and secondary parameters used in the non-LOCA transients.

Response 48:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 49:

The reanalysis of the increased main steam flow transient assumes an initial pressurizer level at the upper limit of 67.5%. Please discuss the consequences if the lower limit of 21% is assumed in this analysis. Will the pressurizer be emptied much earlier in the sequence of event and cause loss of pressure control to RCS?

Response 49:

The higher initial pressurizer level was selected to increase the rate of pressure decrease prior to and during the trip sequence. This results from the cooldown related contraction of the RCS acting upon a smaller steam space. This was done to maximize core thermal margin degradation and increase the potential for DNB.

As seen in PUR Figure 2.13.1.1.3-6, the simulation already predicts emptying of the pressurizer. Voiding in the upper head region of the reactor vessel does develop for a period of time. Thus, the complications of RCS pressure control are already simulated.

Question 50:

Tables 2.6-3 through 2.6-7 listed nuclear steam supply system design transients for Waterford 3. Please confirm that these design transients are applicable for the current core power level conditions and that they are unchanged for the EPU conditions.

Response 50:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 51:

Please confirm that only safety grade systems and components are credited in the reanalysis of all transients and accidents in your EPU report for WSES 3.

Response 51:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 52:

Provide a more detailed rationale for your selection of initial plant conditions for each transient analyzed to achieve the most conservative results.

Response 52:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 53:

Please discuss the significance of assuming an initial power of 1 MWt for the analysis of an inadvertent opening of a steam generator ADV.

Response 53:

The inadvertent opening of a steam generator atmospheric dump valve (IOSGADV) is analyzed in two ways. The first analysis of the IOSGADV is included in the spectrum of excess steam demand events. For this analysis, the thermal margin degradation associated with the excess steam demand events from full power conditions is quantified, and it is ensured that the monitoring system (COLSS) in combination with the action of the reactor protection system is adequate to provide protection. In this context, the excess load imposed by the IOSGADV is less severe than the excess loads that may be imposed by the turbine.

The second analysis of the IOSGADV involves the examination of the impact of the positive reactivity that is added to the core due to the IOSGADV and the resultant cooldown of the RCS. For this event scenario, there is more secondary system mass in the steam generators at low power than at full power conditions. Thus, the potential overcooling of the RCS and amount of positive reactivity addition at low power initial conditions presents the greatest challenge.

The selection of 1 MWt is merely a value which results in the NSSS simulation initializing with maximum steam generator secondary inventory. Other "zero power" values could just as well have been used to maximize the positive reactivity addition associated with the IOSGADV.

Question 54:

To support the results of the reanalysis for the increased main steam flow with LOOP, and sheared shaft with LOOP, please provide the calculated amount of fuel pins with their minimum DNBR (MDNBR) below the allowable MDNBR of 1.26 in each event analyzed. Compare the amount of fuel failure with the acceptance criteria for these events.

Response 54:

The method of Statistical Convolution is being used to predict fuel failure due to DNB for the Waterford 3, 3716 MWt, pretrip power excursion steam line break event. This method of calculating fuel failures was approved in CENPD-183-A, "Loss of Flow, C-E Methods for Loss of Flow Analysis".

In this method, the total number of fuel rods in violation of the DNBR SAFDL is not determined. Rather the number of fuel rods in a given DNBR interval is multiplied by the probability of actually being in DNB for the value of DNBR. This is summed over all DNBR intervals and the total number of fuel pins actually expected to be in DNB is obtained. The conservative assumption is still made that all pins predicted to actually experience DNB are assumed to experience clad failure for the radiological consequences calculation.

There are no specific fuel failure acceptance limits for the seized rotor or excess load with loss of alternating current (LOAC) events. The radiological consequences calculations performed for EPU show that up to 15% of the fuel pins as a result of the sheared shaft and up to 8% of the fuel pins as a result of an excess load with LOAC could experience clad damage and still meet the acceptance criteria. The reload process will ensure the predicted fuel failure for any of the reload cores will remain below these limits by the performance of a cycle specific fuel failure calculation.

Both the use of statistical convolution for all events and the calculation of the fuel failure limits based upon the radiological consequences acceptance criteria were described to the NRC in Entergy letter W3F1-2002-0106, December 16, 2002. Included in that letter are citations of NRC approval of the methods on other dockets for all of the events for which Waterford 3 EPU is using those methods.

The minimum DNBR of the peak pin in the core for the sheared shaft event was 1.0698. For the excess load with LOAC, the minimum DNBR of the peak pin in the core was 1.057.

The uprate project examined several expected power uprate core designs. The fuel failure results, via the method of statistical convolution for those possible core designs are contained in Table RAI.54-1. It is seen that the expected fuel failure for uprate cycle designs is significantly below that which corresponds to the radiological consequences limits.

Table RAI.54-1
Sheared Shaft and Excess Load with LOAC
Fuel Failure via Statistical Convolution for Study Core Design

Core Design	Sheared Shaft		Excess Load with LOAC	
	Cycle Fuel Failure	Limit from Radiological Consequences	Cycle Fuel Failure	Limit from Radiological Consequences
Uprate Cycle 14	8.89%	15%	6.03%	8%

Question 55:

To support the reanalysis of the main SLB accident with LOOP, please provide the following: 1) the calculated amount of fuel pins with their MDNBR below the allowable MDNBR of 1.26 for the cases with a break inside containment; and 2) transient curves for the cases with a break outside the containment.

Response 55:

The method of Statistical Convolution is being used to predict fuel failure due to DNB for the Waterford 3, 3716 MWt, pretrip power excursion steam line break event. This method of calculating fuel failures was approved in CENPD-183-A, "Loss of Flow, C-E Methods for Loss of Flow Analysis".

In this method, the total number of fuel rods in violation of the DNBR SAFDL is not determined. Rather the number of fuel rods in a given DNBR interval is multiplied by the probability of actually being in DNB for the value of DNBR. This is summed over all DNBR intervals and the total number of fuel pins actually expected to be in DNB is obtained. The conservative assumption is still made that all pins predicted to actually experience DNB are assumed to experience clad failure for the radiological consequences calculation.

The radiological consequences calculations performed for EPU show that up to 8% of the fuel pins during the pretrip power excursion SLB could experience clad damage and still meet the acceptance criteria. The reload process will ensure the predicted fuel failure for any of the reload cores will remain below this limit by the performance of a cycle specific fuel failure calculation.

Both the use of statistical convolution for all events and the calculation of the fuel failure limits based upon the radiological consequences acceptance criteria were described to the NRC in Entergy letter W3F1-2002-0106, December 16, 2002. Included in that letter are citations of NRC approval of the methods on other dockets for all of the events for which Waterford 3 EPU is using those methods.

The uprate analyses examined several expected uprate core designs. The fuel failure results, via the method of statistical convolution for those possible core designs are contained in Table RAI.55-1

Table RAI.55-1
Pretrip Power Excursion Steam Line Break
Fuel Failure via Statistical Convolution for Study Core Design

Core Design	Inside Containment Break	Outside Containment Break
Uprate Cycle 14	1.078 %	0 %

Thus the expected fuel failure is expected to be well under the 8% radiological consequences based value submitted in the power uprate report.

Transient curves for the cases with a break outside the containment are presented in Figures RAI-55-1 through RA-55-7.

Figure RAI-55-1
Outside Containment Steam Line Break, Pretrip Power Excursion
Core Power vs. Time

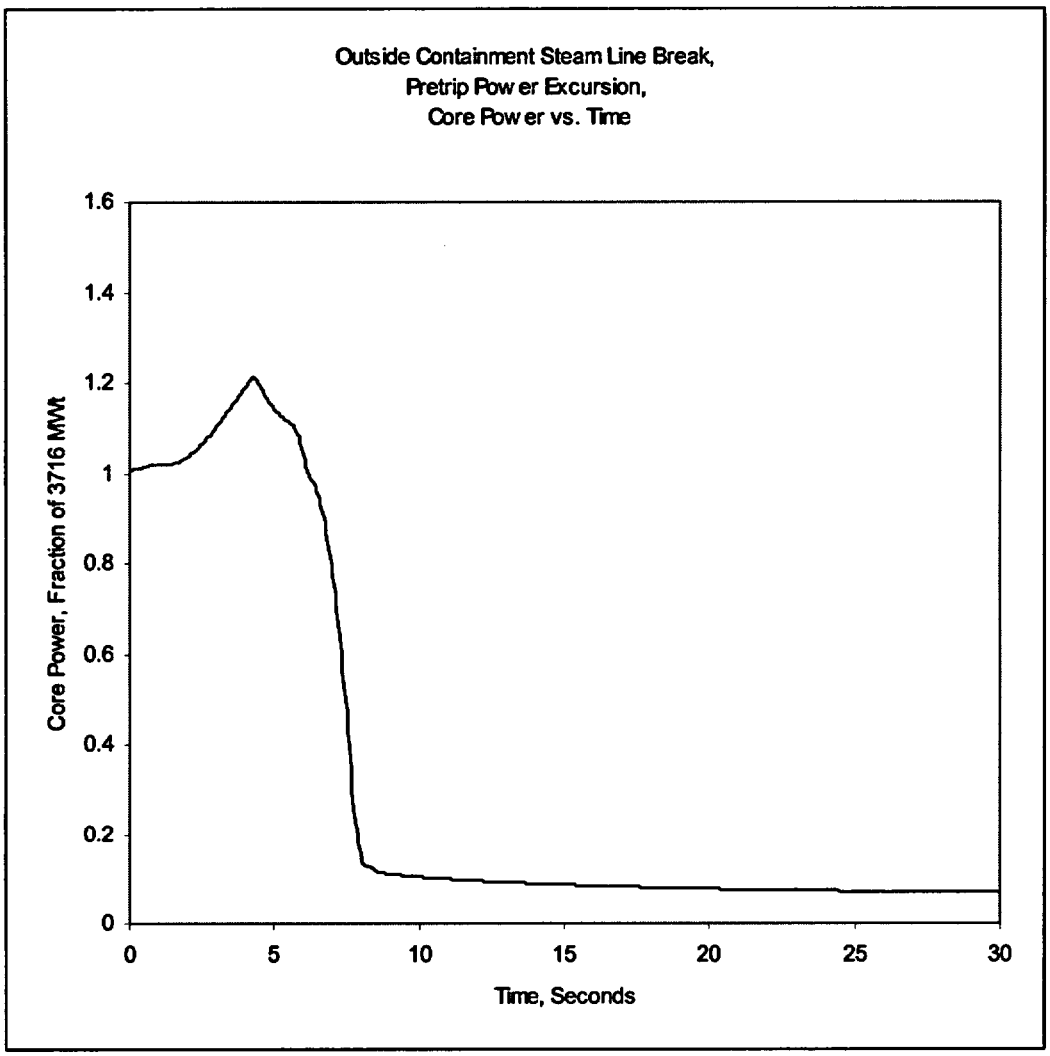


Figure RAI-55-2
Outside Containment Steam Line Break, Pretrip Power Excursion
Core Heat Flux vs. Time

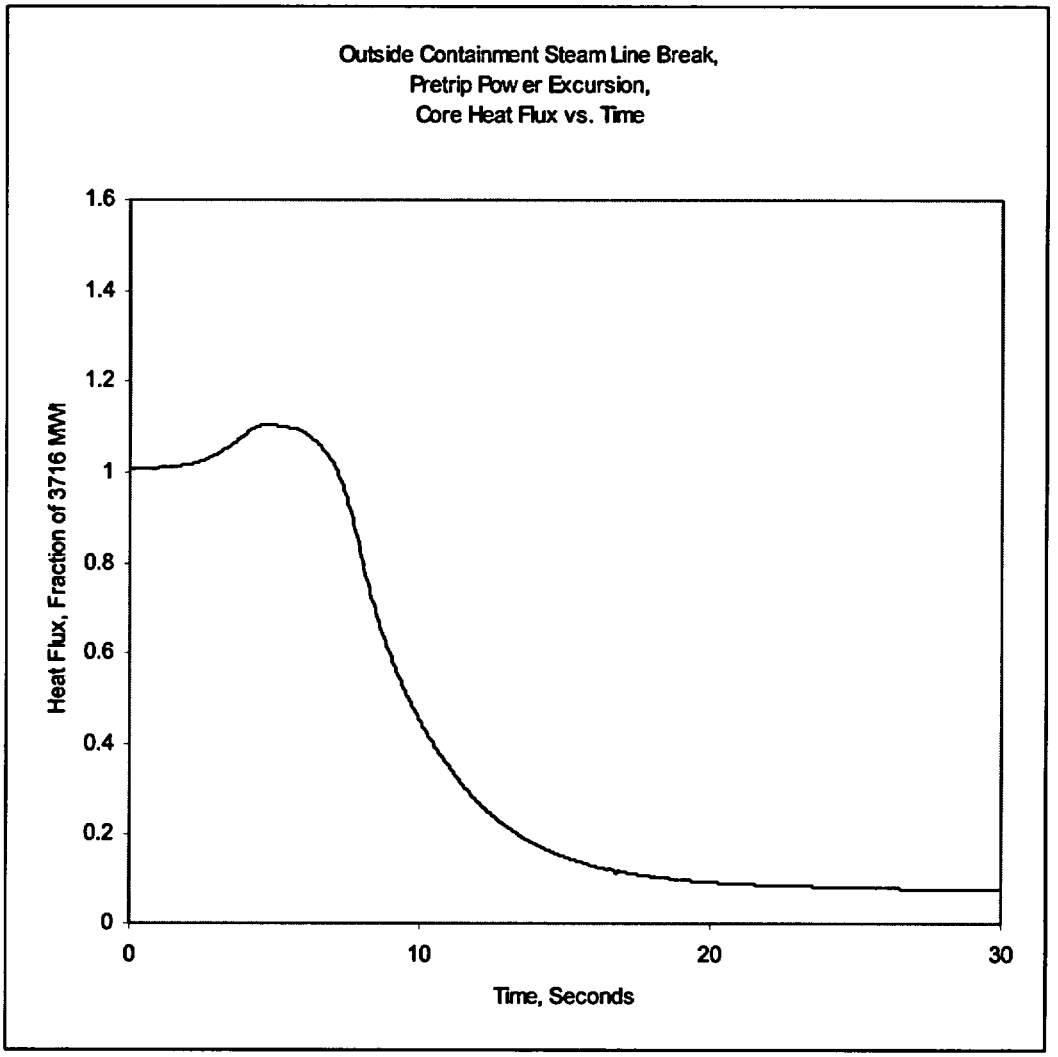


Figure RAI-55-3
Outside Containment Steam Line Break, Pretrip Power Excursion
RCS Temperatures vs. Time

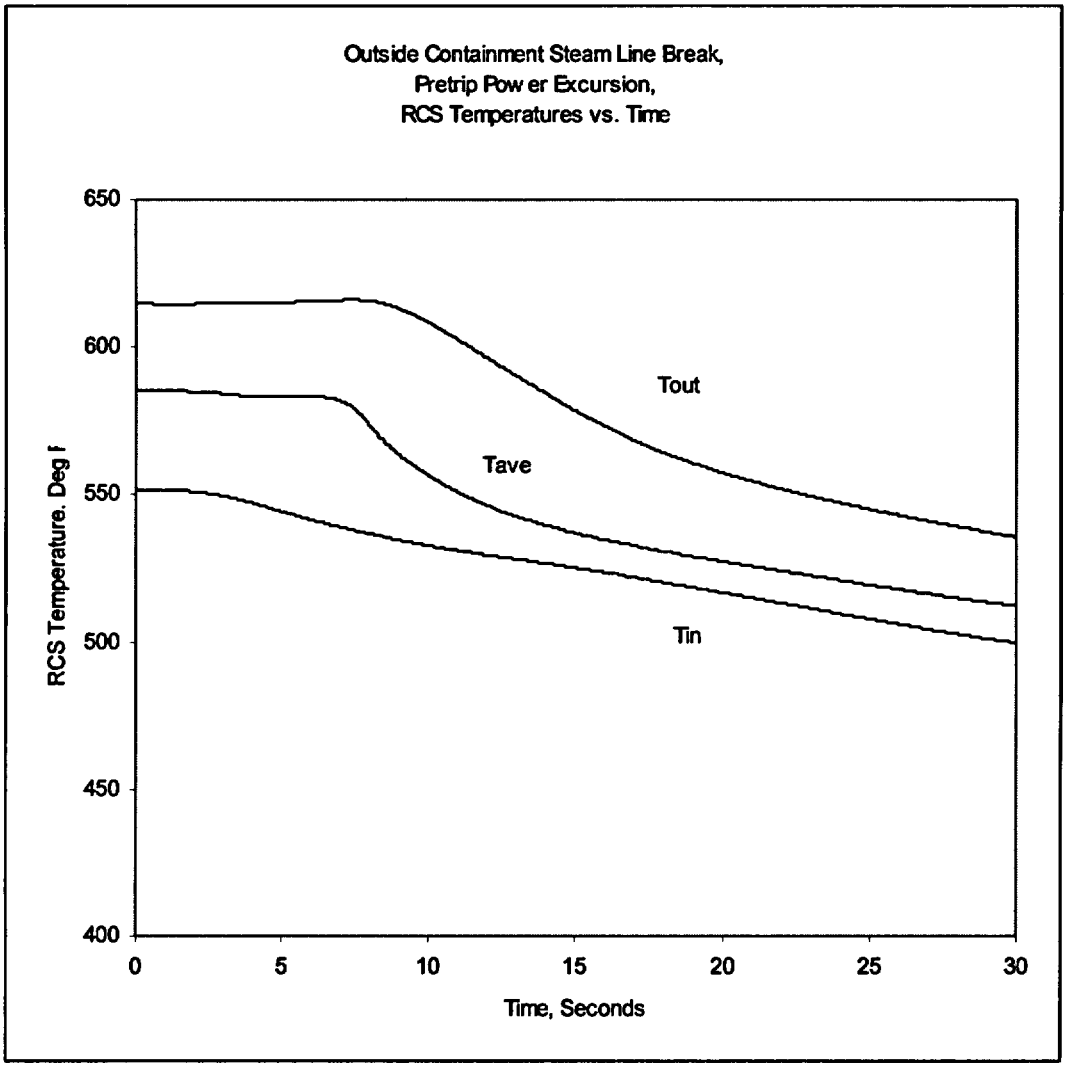


Figure RAI-55-4
Outside Containment Steam Line Break, Pretrip Power Excursion
RCS Pressure vs. Time

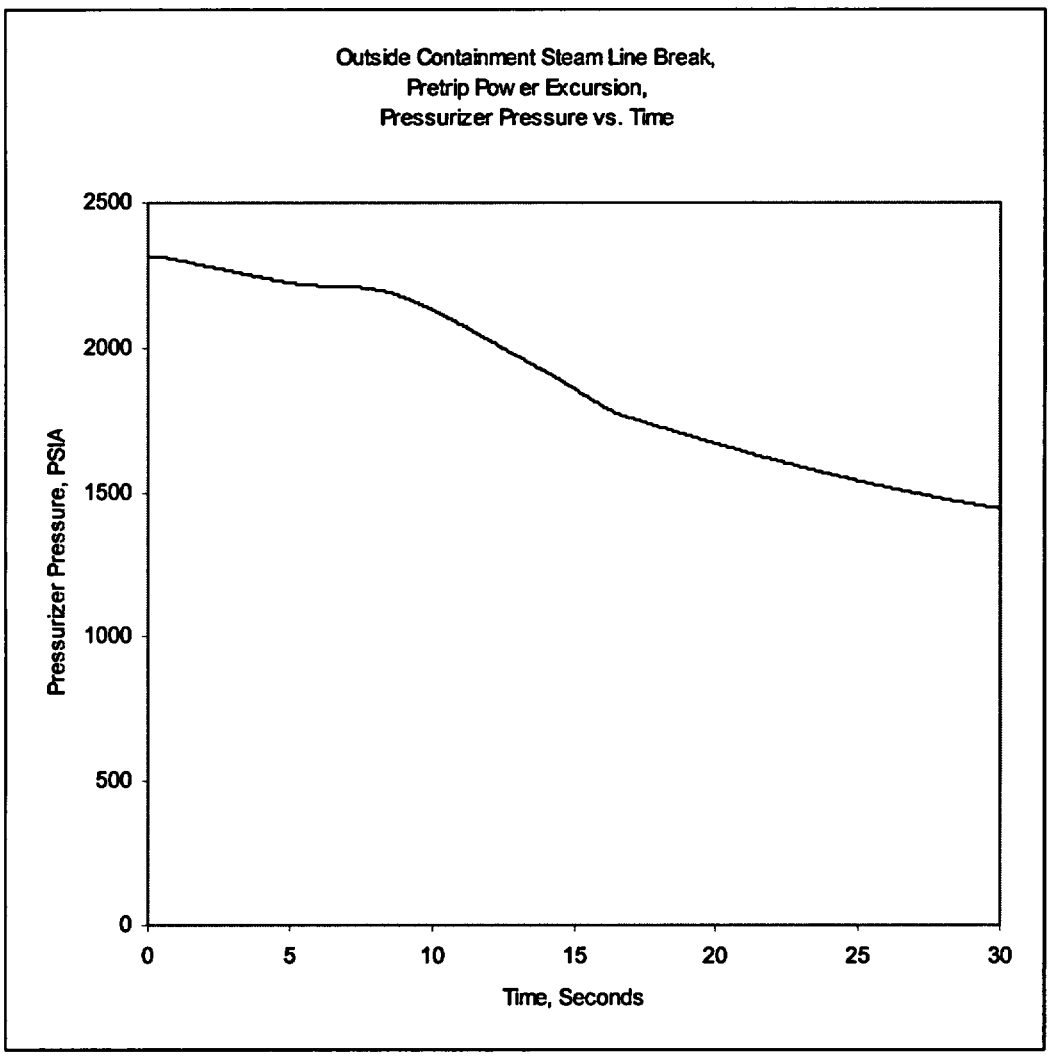


Figure RAI-55-5
Outside Containment Steam Line Break, Pretrip Power Excursion
Reactivity vs. Time

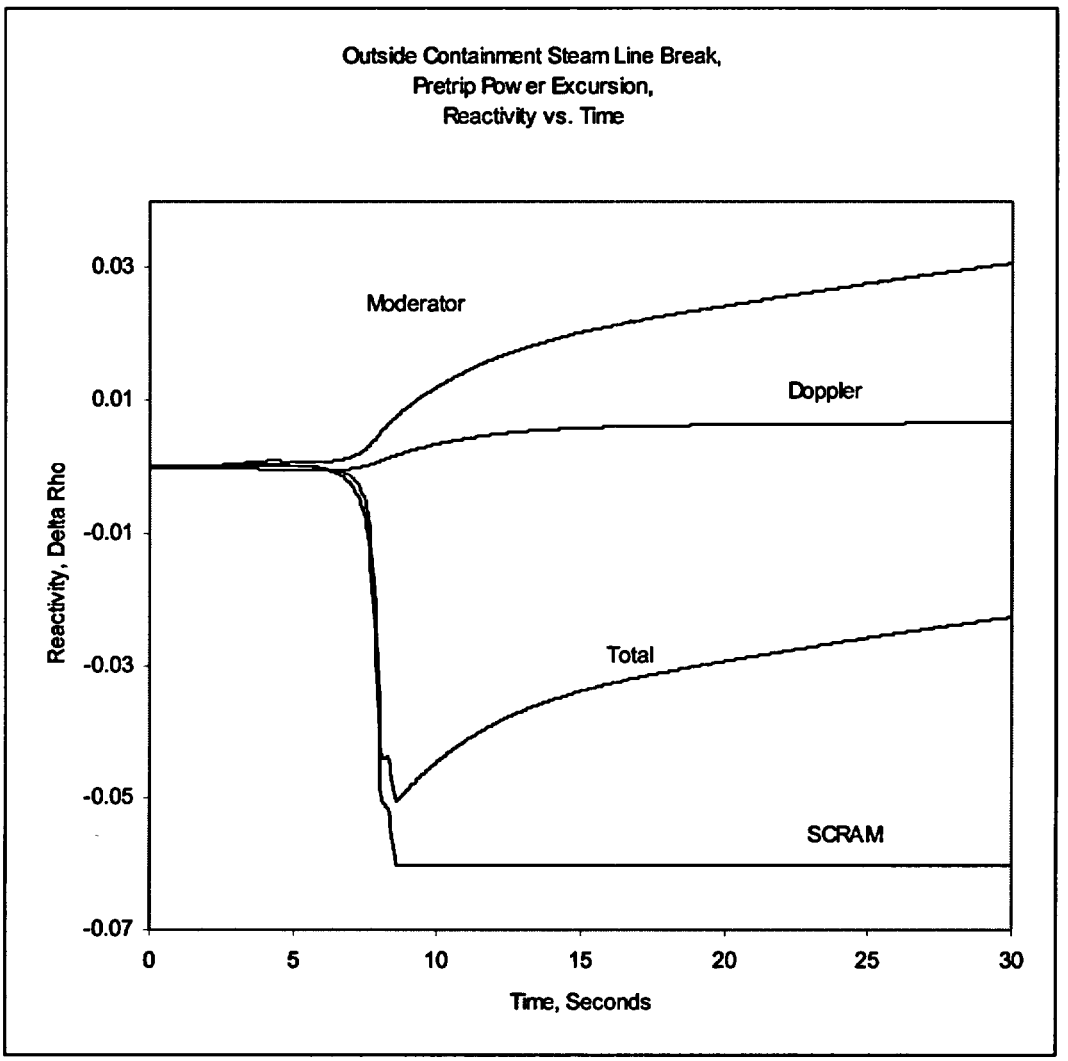


Figure RAI-55-6
Outside Containment Steam Line Break, Pretrip Power Excursion
Steam Generator Pressure vs. Time

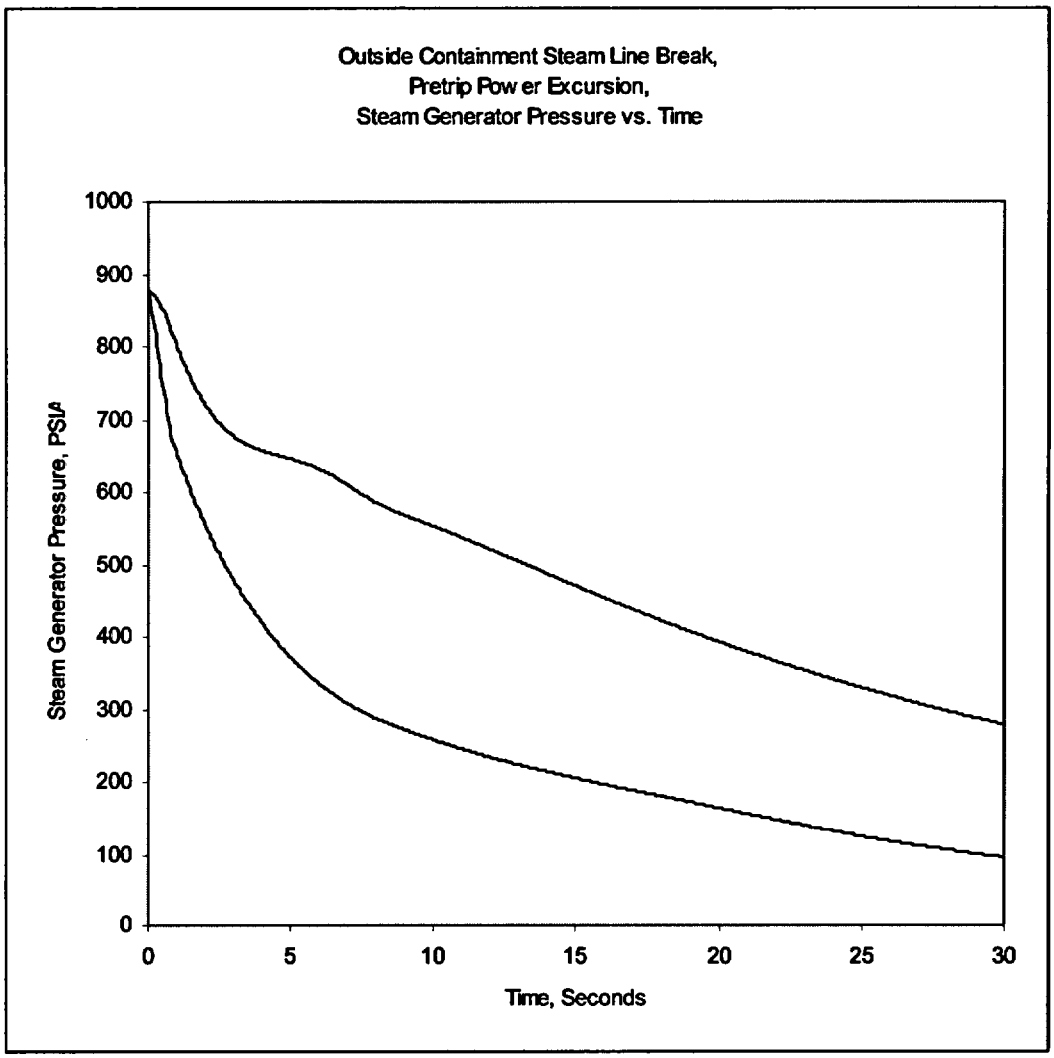
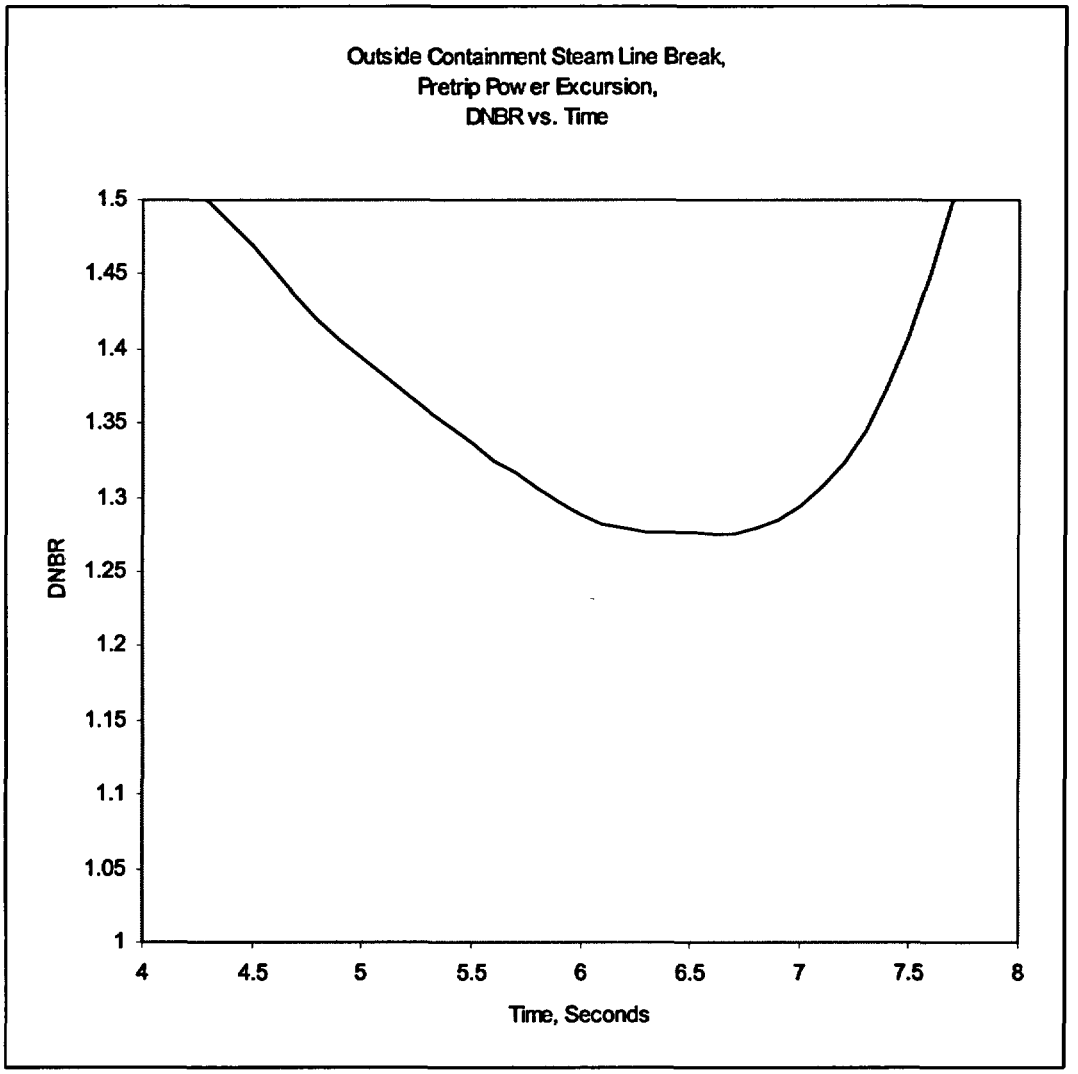


Figure RAI-55-7
Outside Containment Steam Line Break, Pretrip Power Excursion
DNBR vs. Time



Question 56:

For the loss of condenser vacuum transient, please provide the following: 1) the sequence of events for the peak primary pressure case and the peak secondary pressure case; and 2) a separate set of transient curves for each case analyzed.

Response 56:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 57:

Discuss why the assumed break sizes for a main feedwater line break (MFLB) accident is different from that in the current licensing analyses. Provide a discussion of the break size assumed for a large MFLB relative to the double-ended break of a main feedwater pipe.

Response 57:

The methodology used to analyze the feedwater line break for the Waterford 3 extended power uprate, Section 2.13.2.3.1, is consistent with the methodology described in CESSAR Appendix 15B and is the same as the current Waterford 3 licensing basis. Important points of this methodology are:

- Instantaneous loss of feedwater flow to both steam generators at the time of the break.
- Adjusting initial conditions to obtain nearly simultaneous action of the High Pressurizer Pressure Trip and the assumed Low Steam Generator Level trip.
- The fluid leaving the steam generator through the break is saturated liquid. No transition to steam blowdown occurs prior to trip.
- The large feedwater line break assumes that reduction of the affected steam generator heat transfer area doesn't begin until the SG is essentially empty (9,000 Lbm in the analysis versus the expected value of ~131,000 Lbm for Waterford 3).
- The large feedwater line break delays the crediting of the Low Steam Generator Level trip in the affected steam generator until the steam generator is essentially empty (9,000 Lbm in the analysis versus the expected value of >95,000 Lbm for Waterford 3).
- The 'nearly empty' assumption of the large feedwater line break event is too conservative for the small feedwater line break. Therefore, the assumed mass for the onset of the heat transfer reduction (21,000 Lbm in the analysis versus the expected value of ~131,000 Lbm for Waterford 3) and the generation of the Low Steam Generator Level trip is higher (21,000 Lbm in the analysis versus the expected value of >95,000 Lbm for Waterford 3)
- The small feedwater line break spans the spectrum of break sizes from 0.0 to 0.2 Ft^2.
- The large feedwater line break spans the spectrum of break sizes from 0.0 Ft^2 to a double-ended guillotine break.

Variations in the size of the limiting breaks for the current analysis and the power uprate analysis are attributable to the differences in the limits over which the initial conditions are varied due to LCO changes associated with the extended power uprate.

Question 58:

The proposed TS 4.7.1.5.a (surveillance requirements) will change the full closure time of the main steam isolation valve (MSIV) from 4.0 seconds to the analysis value of 8.0 seconds which includes an assumed 1.0 second instrument response time. It is stated in your submittal that a closure time of 4.0 seconds, measured under static test conditions, demonstrates closure under plant operating conditions within the 8.0 seconds assumed in the safety analysis. Please provide the following information: 1) explain how this surveillance requirement could be performed under plant operating conditions assumed in the safety analysis including the instrument response time for the required 8 seconds closure time; and 2) explain why a 4.0 seconds closure time under static test conditions demonstrates closure under plant operating conditions within the 8.0 seconds assumed in the safety analysis.

Response 58:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 59:

The proposed TS 4.7.1.6.a (surveillance requirements) will change the full closure time of the main feedwater isolation valve (MFIV) from 5.0 seconds to 6.0 seconds to include an instrument response time of 1.0 second. Please explain how this surveillance requirement could be performed under plant operating conditions assumed in the safety analysis including the 1.0 second instrument response time for the total required 6.0 seconds closure time.

Response 59:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 60:

The proposed TS 3.7.1.1 specifies the maximum allowable power level with one or two main steam safety valves (MSSVs) inoperable. Please discuss why the maximum allowable power level with more than two inoperable MSSVs on any operating steam generator(s) are not specified in Table 3.7-2 of the proposed TS.

Response 60:

The response to this question was provided in Entergy letter W3F1-2004-0061 dated July 28, 2004.

Question 61:

It is stated that the maximum allowable power level with inoperable MSSVs were determined by the results of the loss of condenser vacuum transients. Please provide the resulting peak primary and secondary pressures for the cases with the current power level compared to that for the uprated power level.

Response 61:

Table RAI #61-1 presents a comparison of the peak system pressures occurring for the loss of condenser vacuum with 0, 1 and 2 inoperable main steam safety valves at the current power level and at the extended power uprate power level.

Note that the analysis of the inoperable main steam safety valves considers only a case tuned to maximize secondary system pressure. This scenario has a high initial RCS temperature to maximize initial SG pressure. The peak secondary pressures for all of the inoperable valve combinations remain below the acceptance criteria of 1210 PSIA.

The case for maximum primary system pressure uses a low initial RCS temperature to maximize the primary system mass in the RCS. The resultant initial SG pressure is low. For these cases the increase in secondary system pressure is not sufficient to open the main steam safety valves prior to the time of peak primary system pressure.

Table RAI #61-1

	Current, Rated Thermal Power 3441 MWt			EPU, Rated Thermal Power 3716 MWt		
Inoperable MSSVs	Maximum Allowed Indicated Power	Peak Primary Pressure, PSIA	Peak Secondary Pressure, PSIA	Maximum Allowed Indicated Power	Peak Primary Pressure, PSIA	Peak Secondary Pressure, PSIA
0	100%	2615	1195	100%	2633	1175
1	86.8%	2565	1192	85.3%	2608	1179
2	69.4%	2510	1193	66.7%	2570	1186

Attachment 2 To

W3F1-2004-0068

Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft

rapid flow reduction causes the pressure difference (ΔP) across the affected loop steam generator (SG) to drop below the low flow differential pressure trip setpoint.

The reactor trip produces an automatic turbine trip. Following the turbine trip, there is an assumed failure to transfer to offsite power, thus a loss-of-non-emergency-offsite power (LOOP) is experienced. This results in a loss of power (LOOP) to the remaining RCPs, steam bypass valves (SBVs), the PPCS and the PLCS. The main impact is that the remaining 3 RCPs begin coastdown, resulting in further loss of flow at a time in which the core is already experiencing possible DNB. This makes the immediate DNB situation more adverse.

This evaluation conservatively analyzed LOOP occurring coincident with turbine trip. DNBR degradation is terminated when the mitigating effects of scram CEA insertion dominate the flow coastdown resulting from the combined effects of the initiating seized rotor/sheared shaft event and the subsequent LOOP.

Table 2.13.3.3.1-1 contains the initial conditions and assumptions used for RCP seized rotor/sheared shaft with LOOP event.

2.13.3.3.1.5 Radiological Consequences

The radiological consequences for the RCP seized rotor/sheared shaft were calculated assuming that the radioisotopes in the gas gap of the pins that experience DNB was immediately mixed with the RCS for release. Releases for site boundary doses were calculated accounting for the carry over of activity to the secondary system via SG tube leakage paths.

The allowed fuel failure limit for the RCP seized rotor/sheared shaft was back-calculated to determine the extent of fuel failure which would result in the regulatory limit for the event. Cycle-specific fuel failure evaluations for power uprate cores will be performed to ensure that this fuel failure limit will not be exceeded. For the RCP seized rotor/sheared shaft, the fuel failure limit to meet the SRP Guidance of a small fraction of 10CFR100 was determined to be 8.0% of the fuel pins. 15.0

The radiological consequences resulting from these fuel failure results are:

	2-Hour EAB	8-Hour LPZ
Thyroid	< 30 rem	< 30 rem
Whole Body	< 2.5 rem	< 2.5 rem

Ref W3F1-2004-0035 dated May 7, 2004

2.13.3.3.1.6 Results

The combined impact of a more adverse flow coastdown and a later time to reach the credited RPS trip condition results in the RCP sheared shaft event being more limiting than the RCP seized rotor event. LOOP occurring coincident with turbine trip was determined to be more limiting that an LOOP occurring shortly after turbine trip. Hence, the most limiting event for this analysis (i.e., resulting in more adverse DNBR degradation) is the RCP sheared shaft event with LOOP at time of trip. A sequence of events for this limiting case is shown in Table 2.13.3.3.1-2.

Attachment 3 To

W3F1-2004-0068

Component Cooling Water Piping Design Temperatures

Component Cooling Water Piping Design Temperatures

In Section 2.5.5.3 of Attachment 5 to the November 13, 2003, Extended Power Uprate (EPU) submittal, Entergy Operations, Inc. (Entergy) noted that the heat load during normal shutdown is higher due to the higher decay heat from the fuel in the reactor vessel and the spent fuel pool and that these higher heat loads increase the temperature of the component cooling water system (CCWS) return flow in some of the CCWS piping sections. Entergy committed to evaluate the impact of these higher temperatures for the CCW piping, supports, and components as well as the impact of these higher temperatures on the shutdown cooling heat exchanger room cooler. Appropriate actions will be taken as necessary based on the results of this evaluation. In Attachment 8 to the November 13, 2003, EPU submittal, Entergy committed to complete these actions by the end of refueling outage 13 (i.e., prior to operating at EPU conditions).

During the evaluation of CCW design temperatures for impact as a result of the EPU, it was determined that, post-EPU, CCW shutdown cooling (SDC) outlet header could exceed its design temperature of 175°F. In addition, it was determined that the CCW SDC outlet header could, and has exceeded, the design temperature of 175°F at the current (pre-EPU) power level. This issue has been entered into Entergy's 10 CFR 50 Appendix B corrective action program. Currently a compensatory action is in place requiring Operations to monitor and maintain CCW return temperature from the SDC outlet header to less than or equal to 225°F. The 225°F limit insures acceptable system and plant protection while the final resolution is identified and implemented. This compensatory action is currently incorporated into plant operating procedures.

It is estimated that it will take in excess of 1000 man-hours and vendor support to identify, justify, and implement the final resolution (e.g., re-rate components for a higher design temperature, etc.) for this issue. Due to the increase in scope, this issue may not be resolved prior to EPU implementation in the Spring of 2005 therefore Entergy proposes to revise the commitment made in Section 2.5.5.3 of Attachment 5 and in Attachment 8 to the November 13, 2003, EPU submittal as follows.

> Ensure that the compensatory actions currently in place to protect the CCW design temperature at the outlet of the SDC heat exchanger are adequate, or revised as necessary, to accommodate the impact of the higher decay heat loads from EPU prior to the implementation of the EPU. Maintain adequate compensatory actions in place until the final resolution (e.g., system rerate, etc.) is identified and implemented.

A mark-up of the page from Attachment 8 to the November 13, 2003, EPU submittal is provided on the following page showing the revised commitment.

PUR Section	Commitment	Type (Check One)		Scheduled Completion Date (If Required)
		One-Time Action	Continuing Compliance	
2.5.2.4	Based on this rule change to 10 CFR 50.44, Entergy will be submitting a separate license amendment request to eliminate the Waterford 3 technical specification requirements for combustible gas control in containment. This license amendment request will be submitted by the end of 2003.	X		12/31/03
2.5.3.1	In response to Generic Letter 2003-01, *Control Room Habitability*, Entergy has committed to complete the requested evaluation prior to the end of September 2004. This evaluation will include a validation of the inleakage assumptions made in the dose consequence analyses. The results of this evaluation will determine further appropriate actions, if any, that must be taken to resolve this issue. (Reference commitment A26565)	X		9/30/04
2.5.5.3	~~These higher heat loads will increase the temperature of the CCWS return flow in some of the CCWS piping sections. The impact of these higher temperatures on the CCW piping, supports and components will be evaluated. Also, the impact of these higher temperatures on the shutdown cooling heat exchanger room cooler will be evaluated.~~	X		~~End of RF13~~ NA
2.5.6.1	In order to accept the higher MSSS flows for EPU, the HP turbine steam path will be replaced.	X		End of RF13
2.5.6.1	The feedwater control system setpoints will be modified slightly to increase pump speed at a lower demand.	X		End of RF13
~~2.5.6.1~~	~~The MSR safety valves are undersized for the EPU conditions and will be replaced as required.~~	~~X~~		~~End of RF13~~
2.5.6.2	Measures will be implemented as necessary to prevent potential condenser tube vibration under power uprate conditions.	X		End of RF13

Ref W3F1-2004-0052 dated 7/14/04

Ensure that the compensatory actions currently in place to protect the CCW design temperature at the outlet of the SDC heat exchanger are adequate, or revised as necessary, to accommodate the impact of the higher decay heat loads from EPU prior to the implementation of the EPU. Maintain adequate compensatory actions in place until the final resolution (e.g., system rerate, etc.) is identified and implemented.

Attachment 4 To

W3F1-2004-0068

List of Regulatory Commitments

List of Regulatory Commitments

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

REVISED COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
The response to the final unanswered question (i.e., #20) will be submitted by August 25, 2004.	X		8/25/04
Ensure that the compensatory actions currently in place to protect the CCW design temperature at the outlet of the SDC heat exchanger are adequate, or revised as necessary, to accommodate the impact of the higher decay heat loads from EPU prior to the implementation of the EPU. Maintain adequate compensatory actions in place until the final resolution (e.g., system rerate, etc.) is identified and implemented.	X		NA