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W3F1-2004-0037

Bradford Houston
Director, Nuclear Safety Assurance
Waterford 3

May 12, 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 March 31, 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)"

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 containment analysis. Entergy's response to these eight questions is contained in the Attachment 1 to this letter.

Additionally, Entergy is providing supplemental information regarding reactor coolant system flow as follow-up to information discussed during the February 5, 2004, meeting with the NRC staff. This information is contained in Attachment 2 to this letter.

There are no technical changes proposed. The original no significant hazards consideration included in Reference 1 is not affected by any information contained in this letter. There are no new commitments contained in this letter.

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

A001

I declare under penalty of perjury that the foregoing is true and correct. Executed on May 12, 2004.

Sincerely,

Bradford L Houston

BLH/dbm

Attachments:
1. Response to Request for Additional Information
2. Additional Reactor Coolant System Flow Information

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Attachment 1 To

W3F1-2004-0037

Response to Request for Additional Information

**Response to Request for Additional Information
Related to the Extended Power Uprate**

Question 1:

Verify that all input parameters to the containment peak pressure and temperature (both loss-of-coolant accident (LOCA) and main steam line break (MSLB)), minimum pressure LOCA, environmental qualification (EQ), and subcompartment analyses remain the same as those in the final safety analyses report (FSAR) except for those affected by the power uprate. For example: containment volume, heat sink descriptions, heat exchanger performance, equipment flow rates and flow temperatures, initial relative humidity, refueling water storage pool (RWSP) temperature, ultimate heat sink temperature, etc. Justify any changes made for the power uprate analyses.

Response 1:

Changes in input parameters for the containment peak pressure and temperature calculations in support of extended power uprate (EPU) included the following:

a. SDCHX Performance:
Shutdown cooling heat exchanger (SDCHX) is used to cool the containment spray flow during recirculation mode. The heat exchanger model in GOTHIC uses the primary and secondary convective heat transfer coefficients (HTC) and tube material property to calculate the primary to secondary heat transfer. The analysis currently documented in the FSAR uses constant HTCs. In the EPU analysis, the SDCHX primary and secondary HTCs were calculated as a function of spray (safety injection sump) temperature and were input into GOTHIC as tables that were interpolated as a function of the GOTHIC calculated sump temperature. This change more realistically, but still conservatively, models the SDCHX performance. This change has no impact on MSLB results or the post-LOCA containment peak pressure and temperature, since the peaks occur well before the recirculation actuation which causes containment spray to take suction from the containment safety injection sump. However, it may have a slight impact on containment pressure at 24 hours post-LOCA. Also for conservatism 5% of the SDCHX tubes were assumed to be plugged.

b. Containment Spray Delay Time:
An additional one second delay time has been included in the analysis to account for the impact of the potential presence of a small volume (4 ft^3 assumed) of non-condensable gases in the Containment Spray piping.

c. Mass and Energy Releases:
MSLB mass and energy releases were recalculated for EPU conditions and are different than mass and energy release data currently documented in the FSAR. LOCA mass and energy releases had previously been recalculated based on EPU conditions and had previously been incorporated into the Waterford 3 licensing basis for containment pressure-temperature response (reference Safety Evaluation for Amendment 165 dated July 6, 2000). The LOCA mass and energy release data for EPU are therefore, the same as the mass and energy release data currently documented in the FSAR and previously approved in Amendment 165.

There were no changes to parameters assumed in the analysis for parameters such as containment volume, containment heat sinks, containment spray flow rate and temperature, containment initial pressure and temperature.

Containment parameters assumed for the minimum containment pressure response are discussed in FSAR Section 6.2.1.5. The heat sink data assumed is based on the data in FSAR Table 6.2-7 with additional heat sinks added. Although the data has been slightly rearranged for improved documentation, the heat sink assumptions for EPU are basically equivalent to the information previously specified for pre-EPU ECCS performance analyses. For EPU, an additional steel heat sink of 12,000 ft2 and 0.5 inch effective thickness was assumed for additional conservatism. Also, more conservative assumptions were made (to allow more operating margin to the input assumptions) for minimum RWSP temperature (i.e., containment spray temperature), maximum containment spray flow, and minimum containment temperature and pressure than for the analyses described in the FSAR:

Minimum Containment Pressure Analysis for ECCS

Parameter	Pre-EPU (FSAR)	EPU
Minimum RWSP / spray temperature	55°F	50°F
Maximum spray flow, two pumps	4180 GPM	4500 GPM
Minimum containment temperature	100°F	90°F
Minimum containment pressure	14.375	14.025

Subcompartment pressurization for power uprate is discussed in the response to Question #8.

Question 2:

It appears that the proposed power uprate will use the graded approach to considering instrument uncertainties for the power uprate. Please respond to the following questions concerning the graded approach.

(i) How are the parameters selected which will be subject to the graded approach?

(ii) Branch Technical Position HICB-12, "Guidance on Establishing and Maintaining Instrument Setpoints," Version 7.0, states that the licensee should consider "all known applicable uncertainties regarding setpoint application" when utilizing the graded approach. Recognizing that this position applies to instrument setpoints, nevertheless, justify the fact that the proposed use of the graded approach for containment analysis does not consider uncertainties at all for those parameters included in the graded approach. The containment analysis uses the selected parameters at their nominal values.

(iii) Please describe how the use of the graded approach is consistent with the Waterford 3 technical specifications (TS). For example, the RWSP temperature is listed as a parameter to which the graded approach would be applied. The TS specify a value of 100 °F. The analysis uses a value of 100 °F. How is instrument uncertainty taken into account in this case? Discuss, in general, the relationship between the Waterford 3 TSs and the graded approach used for containment analysis.

(iv) What assurance is there, in applying the graded approach to containment analysis, that the containment design pressure would not be exceeded if the uncertainties were included? The staff does not consider it acceptable to credit the undefined margin between the containment design pressure and the (undefined) ultimate containment failure pressure.

Response 2:

The response below discusses the selection of input parameter values used in the Waterford 3 containment pressurization analysis. Entergy has decided not to specifically address the "graded approach" to instrument uncertainty in this response. Instead, Entergy's response focuses on the acceptability of the containment analysis performed in support of the EPU. The in-depth discussion below is provided to demonstrate the adequacy of the Waterford 3 containment pressurization analyses. The Waterford 3 containment pressurization analysis is an analysis performed using the GOTHIC containment analysis code consistent with the guidance of NUREG-0800, Standard Review Plan, section 6.2, to demonstrate the adequacy of the design basis for the containment structure.

<u>Waterford 3 Input Parameters:</u>
The results of the containment pressurization analysis are dependent on values of input parameters assumed in the analyses. Safety analyses generally assume worst case allowed value for multiple variables (e.g., worst case RCS temperature, pressure, flow; worst case flows and timing for mitigating systems; RWSP volumes and temperature; EFW temperature; etc....) Also there are many cases where mutually exclusive conservatisms are applied in the analysis for simplicity; for example, GOTHIC containment analyses for maximum Safety Injection (SI) flow cases assume both trains of SI are running, but also for containment cooling failure cases inherently are assuming single failure of an EDG resulting in only one Containment Fan Cooler and one Containment Spray pump responding to the event. Waterford 3 uses a suitably conservative set of input parameters in the analysis, per SRP Section 6.2.1 paragraph 4.1, chosen to maximize the containment temperature and pressure response. The containment parameters used are not nominal values, as stated in the question. Specific inputs include:

- An initial containment temperature of 120°F, corresponding to the maximum value per Technical Specifications. The allowed operating range for this parameter is 90°F to 120°F. A nominal value for temperature, based on historical data, would be slightly below 110°F.
- An initial containment pressure of 1.0 psig. The allowed range for this parameter, per Technical Specifications, is 14.275 psia to 27" w.g. (0.974 psig). A nominal value for the containment pressure, based on historical data, would be about 15.0 psia (i.e., about 0.3 psig).
- A refueling water storage pool (RWSP) temperature of 100°F, corresponding to the maximum value per Technical Specifications. The allowed operating range for this parameter is 55°F to 100°F. The range for the nominal value for this parameter, based on historical data, would be between 65°F and 80°F.
- A component cooling water (CCW) flow rate of 1100 GPM to containment fan coolers is assumed, corresponding to the minimum Technical Specification flow rate of 1200 GPM with an assumed allowance of 100 GPM for flow measurement uncertainty. This assumed

allowance is larger than the value required based on plant calculations for the uncertainty allowance for CCW flow surveillances.

- A containment riser level of 149.5 ft is assumed, corresponding to the minimum value per Technical Specification Surveillance Requirement 4.6.2.1a.
- A RWSP available volume of 383,000 gallons is assumed. This volume explicitly (i.e., conservatively) accounts for instrument measurement uncertainty in the RWSP volume when recirculation is initiated, as well as an allowance for uncertainty in measurement of the initial RWSP level.
- The maximum design CCW temperature of 115°F is assumed throughout the event; heatup of the CCW system from its initial temperature (approximately 90 °F) at the start of the accident is conservatively ignored.
- A minimum containment volume of 2,677,000 ft^3 is assumed. Nominal volume is considered to be approximately 2,680,000 ft^3.
- Conservatively large fouling factors for heat exchangers (shutdown cooling heat exchanger and containment fan coolers). Waterford 3 complies with Generic Letter (GL) 89-13 in that Waterford 3 has a program to evaluate the performance of the service water heat exchangers. The SDCHX and the containment fan coolers are currently in the Waterford 3 GL 89-13 heat exchanger test program.
- Minimum containment spray flow of 1750 GPM is assumed. Nominal spray flow is considered to be 2000 GPM.
- A containment spray actuation signal (CSAS) setpoint of 5.0 psig is assumed in the analyses. The plant value for this setpoint is 17.7 psia (3.0 psig), as documented in Technical Specification Table 3.3-4. This is an Engineered Safety Features Actuation system actuation trip setpoint per Technical Specification 3.3.2

Conservative mass and energy release calculations are provided to drive the containment response calculations. FSAR sections 6.2.1.3 and 6.2.1.4 discuss the mass and energy release methodologies for loss of coolant accident (LOCA) and main steam line break (MSLB). The peak containment pressure is higher for MSLB than for LOCA, as documented in Section 2.5 of the PUR. Amongst the conservative assumptions in the MSLB mass and energy release calculations are:

- Maximum RCS cold leg temperature of 552°F is assumed. This is based on the maximum value of 549°F per the proposed EPU Technical Specifications, with a 3°F instrument uncertainty. The nominal value for cold leg temperature is 543°F.
- Maximum RCS pressurizer pressure of 2310 psia, corresponding to the maximum value of 2275 psia per Technical Specifications, with a 35 psi instrument uncertainty. Nominal value for pressurizer pressure is 2250 psia.
- A conservatively large RCS flow of 120% of Technical Specification minimum flow; actual flow is slightly less than 110% of this value. Flow measurement uncertainty is less than 5%.
- An additional conservatism in MSLB analysis is that fan cooler performance ignores the presence of superheated conditions in the containment.
- Hot Zero Power (HZP) MSLB cases assume the mutually exclusive conservatisms of SG inventory based on the minimum Tcold of 533°F (536°F Technical Specification minimum

and 3°F allowance for instrument uncertainty) combined with the assumption of a nominal 541°F HZP Tcold for the remainder of the analysis.

LOCA mass and energy release calculations also assume the Technical Specification maximum values for containment temperature and pressure and for RWSP temperature. A safety injection tank (SIT) pressure of 685 psia is assumed, corresponding to the maximum value permitted per Technical Specifications. A significant conservatism in the Waterford 3 mass and energy release calculation is the assumption of the least resistance K-factor for the SIT's.

The LOCA containment pressure / temperature response analyses using the GOTHIC code assume a loss of offsite power and availability of only one AC power train, minimizing the capacity of the heat removal systems. However, the mass and energy release calculation which is an input to the GOTHIC analysis maximizes the release rates by assuming maximum safety injection flow rates (including the case where both trains of AC power are available). Note that the worst case containment peak pressure due to LOCA is for the hot leg break, for which the time of the peak pressure is during the initial blowdown phase before any safety injection flow reaches the RCS.

Waterford 3 uses assumptions which are acceptably conservative. Use of worst-case values (i.e., extreme values allowed per Technical Specifications) for various input parameters without an additional explicit penalty for instrumentation uncertainty is the established licensing basis for Waterford 3. This is similar to other plants and, in consideration of the other conservatisms discussed, is consistent with the SRP guidance to use acceptably conservative input parameters.

Differences between Setpoints and Initial Conditions:
Setpoints, as defined in ISA-S67.04, are "a predetermined value for the actuation of the final actuation device to initiate protective action." The initial conditions used in the containment pressure analysis do not meet this definition. For containment analyses, NUREG-0800 does not require the assumption or allocation of specific instrument uncertainties in the analyses. Waterford 3's selection of initial conditions for containment pressurization analyses is consistent with the NUREG-0800 guidance to use a suitably conservative set of assumptions for initial operating conditions. This conservatism is enhanced by the combination of multiple parameter initial conditions assumed to be at their worse case value simultaneously.

Margin of safety
Inherent in the consideration of adequacy of safety analyses is the "margin of safety," or the difference between the acceptance limit and the ultimate failure point. The margin of safety may have both quantifiable and non-quantifiable components. While Entergy Operations, Inc. (Entergy) considers that the selection of input parameter values for Waterford 3 containment analysis is consistent with regulatory requirements and the SRP guidance without the consideration of the "Margin of Safety" above the containment pressure design limit, consideration of the "Margin of Safety" provides additional robustness of the analyses in demonstrating the adequacy of the containment design against overpressurization.

While not required to support the acceptability of the analyses, consideration of the "Margin of Safety" would be consistent with the NRC's increasing focus on risk-informed regulation. All available information should be used in determining safety significance. As stated in Entergy's

October 6, 2000, letter to the NRC, Entergy did assess the impact of instrument uncertainty on containment integrity during an accident. The assessment concluded that the instrument uncertainty in the identified parameters was not safety significant. As stated in Entergy's earlier October 18, 1999, submittal to the NRC, there is large margin available between the containment pressure limit of 44 psig and the actual failure pressure of containment. Specifically, the uncertainties associated with instrument measurement uncertainty are quite small compared to the "margin of safety."

This provides additional confirmation of the adequacy of the Waterford 3 containment pressurization analyses to demonstrate acceptable design against containment overpressurization and thus the integrity of this fission product barrier.

<u>Sensitivity analyses:</u>
Sensitivity analyses to study the impact of instrument uncertainty upon calculated containment performance have been performed previously. Similar sensitivity studies were performed for EPU conditions. Conservative measurement uncertainties were assumed for containment pressure and temperature, spray riser level, RWSP temperature, and CCW temperature. Note conservative treatment of measurement uncertainties are already accounted for in CCW flow and in the RCS temperature and pressure initial condition assumptions for Mass and Energy releases. The uncertainties assumed are considered equal to or greater than that corresponding to actual plant measurement uncertainties. For example, this sensitivity study assumed a 7 foot uncertainty in riser level, whereas the actual measurement uncertainty is calculated to be ~4.9 foot.

Consideration of these uncertainties on top of the assumption of initial conditions chosen to maximize containment response resulted in minor increases in calculated peak pressures (0.30 psi for LOCA and 0.71 psi for MSLB). Note this is based upon applying these uncertainties to all five parameters at once. A proper statistical treatment, consistent with the treatment of environmental factors in setpoint methodology calculations, would have considered the impact of each individual variable and applied a root sum of squares treatment to the resulting differences in pressures. This would have further reduced these small variabilities.

These uncertainties are considered two-sided 2σ uncertainties, for which there is only a 2.5% probability of exceeding the uncertainty value. Thus, when such uncertainties are applied to five parameters (CCW temperature, RCS temperature, RCS pressure, spray riser level, RWSP temperature), there is only a $(.025)^5 = 9.8\times10^{-9}$ probability of these parameters all being at the extreme value corresponding to instrument uncertainty. This is on top of the already low probability that the parameters will be at the limiting values allowed per Technical Specifications and/or plant procedures.

<u>Conclusion:</u>
The assumptions that critical parameters are all at worst case extreme of their operating range is such a low probability situation that it is <u>not credible</u> to further postulate that all these parameters are at the worst case extreme plus instrument uncertainty. Due to the statistical nature of the derivation of instrument uncertainty, it results in an arbitrarily over conservative analysis to explicitly apply uncertainty to the multiple input parameters in the analysis. It would result in disproportionate restrictions on plant operations and increase in design basis burden for the negligible level of additional protection associated with explicit treatment of instrument

uncertainty for all input parameters. For example, there would be no gain in safety if LOCA analyses had to explicitly account for pressurizer level measurement uncertainty.

- The Waterford 3 containment pressure analyses assume the limiting values (not nominal values) allowed per Technical Specifications for applicable parameters such as containment pressure and temperature and RWSP temperature. This adds sufficient conservatism to the analysis results.
- Initial conditions for safety analyses, such as containment temperature and pressure or RWSP temperature, are not setpoints. There is no automatic actuation of equipment to mitigate events at these parameter's initial conditions.
- The probability that all parameter values will be at the extreme values corresponding to explicitly treated instrument uncertainties is so low as to not be credible.

Thus, the selection of assumed initial operating conditions for Waterford 3 containment pressurization analyses is consistent with regulatory guidance and is consistent with the NUREG-0800 guidance to be suitably conservative. Consideration of the "margin of safety" with regard to containment pressure provides additional justification for the acceptability of the Waterford 3 analyses in demonstrating the adequacy of the containment.

Question 3:

The version of GOTHIC has been changed for the analyses in this submittal from GOTHIC 5.0 to GOTHIC 7.0.

(i) Please verify that the use of GOTHIC 7.0 is consistent with the conditions discussed in an NRC letter to Nuclear Management Company dated September 29, 2003, on the Kewaunee docket (NRC ADAMS Accession Number ML032681050).

(ii) Has a determination been made, in accordance with Title 10 of the Code of Federal Regulations (10 CFR), Section 50.59, that prior NRC review and approval of the use of GOTHIC 7.0 for power uprate calculations is not required? Please specify the specific criteria of 10 CFR 50.59 which are satisfied to support this conclusion.

Response 3(i):

The restrictions identified in the above NRC letter to Nuclear Management Company and their applicability to Waterford 3 are:

- The height effect scaling factor λ_h applied to the heat and mass transfer analogy, shall not be used for Kewaunee licensing calculations:

  Waterford 3 containment analyses do not apply the height effect scaling factor λ_h in the heat transfer calculation in the containment.

- The Gido-Koestel (G-K) correlation shall not be used for Kewaunee licensing calculations.

  The Gido-Koestel (G-K) correlation is not used in Waterford 3 containment analyses.

- The inclusion of mist in the mist diffusion layer model (MDLM) shall not be used for Kewaunee licensing calculations.

  The mist diffusion layer model (MDLM) is not used in Waterford 3 containment analyses.

Therefore, use of GOTHIC 7.0 at Waterford is consistent with the conditions discussed in NRC letter to the Nuclear Management Company dated September 29, 2003.

Response 3(ii):

GOTHIC 7.0 has not previously been used at Waterford 3, and thus no GOTHIC 7.0 applications have been subject to a 50.59 review at Waterford 3. The GOTHIC 7.0 results have been submitted in support of EPU, therefore prior NRC approval for the GOTHIC 7.0 results will be obtained with the approval of the EPU prior to GOTHIC 7.0's use for FSAR Section 6.2 containment pressurization analysis at Waterford 3.

The Waterford 3 containment analyses are being performed in the same manner with GOTHIC 7.0 as previously and no new code features or models are being applied. For confirmation, an informal comparison of GOTHIC 5.0c and GOTHIC 7.0 results were made. The comparison showed that GOTHIC 7.0 results in slightly higher peak pressure (approximately 0.35 psi) for the limiting containment pressure main steam line break event; higher peak pressure is conservative for assessing methodology changes. Thus, it is concluded that use of GOTHIC 7.0 for this application in association with the Waterford 3 EPU does not result in a departure from the current method of evaluation.

Question 4:

Verify that the same assumptions are made regarding the use of 8 percent reevaporation as in the FSAR (Page 6.2-8).

Response 4:

Similar to the case documented in the current FSAR, the 8% re-evaporation is not considered for the containment response to MSLB event.

Question 5:

Verify that the MSLB break area is adjusted to provide dry steam to the containment, as described in the FSAR (Page 6.2-8).

Response 5:

The break areas for MSLB event were selected as the largest break area that resulted in a pure (dry) steam blowdown.

Question 6:

Verify that the methods and assumptions for calculating the EQ envelope have not changed from those described in the FSAR.

Response 6:

Waterford 3 FSAR Section 6.2.1.1.3 discusses the environmental design of equipment. The methods used to calculate the EQ envelope have not changed. For consistency with the containment peak pressure and temperature analyses, the EPU calculations of the MSLB EQ envelope assumes a 0% revaporization, which is conservative with respect to the 8% revaporization allowed for this analysis. The mass and energy release input was updated, along with other minor changes documented in the response to Question #1 above.

Question 7:

Verify that net pump suction head (NPSH) calculations for the emergency core cooling system pumps and containment spray pumps have been revised and that the results are acceptable. Have the required NPSH values ($NPSH_R$) of these pumps been revised?

Response 7:

The NPSH calculations for the emergency core cooling system (ECCS) pumps and the containment spray pumps do not require a revision to support the extended power uprate (EPU) of Waterford 3. No change is required in the performance from the ECCS or containment spray pumps or the associated systems to meet any of the safety analysis acceptance criteria. EPU does not require any change to the data supporting the NPSH calculations as described in Final Safety Analysis Report (FSAR) Section 6.3.2.2.2.3.

Question 8:

Please specify any differences from the FSAR in the analytic methods and assumptions used to perform the subcompartment analyses.

Response 8:

Waterford 3 subcompartment analyses are discussed in FSAR Section 6.2.1.2 and, for EPU, in PUR Section 2.5.2.2. Mass and energy releases for power uprate conditions were generated using the CEFLASH-4A code. CEFLASH-4A was also used to generate the mass and energy releases for the original licensing basis analyses. As stated in the PUR, of the limiting break sizes, the mass and energy releases for EPU conditions were higher only for the 350 in^2 RCS discharge leg break. This break resulted in the limiting pressurization of the reactor cavity. For pre-EPU conditions, the resulting peak pressure load on the reactor cavity wall is 130.3 psi, as documented in FSAR Table 6.2-2. The design limit is 240 psid, per FSAR Table 6.2-3. Due to the effort involved in reconstructing the detailed subcompartment model and the large existing margin, the impact of the increased mass and energy release for the 350 in^2 RCS discharge leg break was determined by developing and applying a conservative scaling factor to the original RELAP analyses. For this break, the mass and energy releases are approximately 7% higher during the first second of the event, when the peak subcompartment pressure occurs. The impact of the higher mass and energy release were evaluated using a simple equilibrium model, a simple non-equilibrium model using the GOTHIC code, and interpolation between break size in original design analyses. These methods led to a conservative engineering judgement that that calculated increase in subcompartment pressurization would be less than a 5.8% increase, which is small compared to the available 84% margin in the original detailed RELAP analyses. RELAP (supplemented as described here) remains the methodology of record for the subcompartment analyses.

Attachment 2 To

W3F1-2004-0037

Additional Reactor Coolant System Flow Information

Additional Reactor Coolant System Flow Information

One item of discussion at the February 5, 2004, meeting with the NRC was the change in the reactor coolant system (RCS) flow assumption for the Waterford Steam Electric Station, Unit 3 (Waterford 3) operating point. As discussed in section 1.2 of the Power Uprate Report (PUR) (attachment 5 of the November 13, 2003, submittal), nominal temperature conditions are based upon a nominal RCS flow of 110% of the minimum design flow. Previously, the nominal flow had been assumed to be 107%. However, due to the condition of Waterford 3 operation at greater than rated thermal power (Waterford 3 Licensee Event Report (LER) 2002-006) RCS flow rates had been underestimated by surveillance procedures in Cycle 11 and previous fuel cycles. This resulted in the incorrect identification of 107% nominal RCS flow as the basis for the Cycle 12 operating point calculation, performed for Appendix K margin recapture power uprate.

RCS flow data was subsequently reviewed by Waterford 3 engineering. Based on the review, a flow rate of 435,600 gpm (110% of the 396,000 gpm design value) was selected for use in the Operating Point calculation for 3716 MWt Extended Power Uprate. RCS flow surveillances throughout Cycle 12 averaged 109.8% of 396,000, providing support for this decision. The RCS flow data is consistent with the data seen in Cycle 6 and prior fuel cycles, prior to the implementation of the Leading Edge Flow Monitor (LEFM) for measurement of feedwater flow which contributed to errors leading to operation of several cycles at slightly greater than rated thermal power. When corrected for power, the RCS flow data demonstrated no significant change in actual flow between Cycle 11 and Cycle 12. Review of reactor coolant pump differential pressure data indicated only a 0.1% increase in volumetric flow between Cycle 11 and Cycle 12; this is in the range of normal data scatter.

Waterford 3 safety analyses are based upon minimum or maximum RCS flow rates, as appropriate for the analysis. Minimum RCS flow, per Technical Specification 3.2.5, is 148 million pounds per hour. A maximum RCS flow of at least 115% of this value is used for analysis which conservatively use a maximum RCS flow.