From: "MILLER, D BRYAN" <dmill14@entergy.com>
To: "'Thomas Alexion'" <TWA@nrc.gov>, "'KALYANAM, N. KALY'" <nxk@nrc.gov>, "'agh1@nrc.gov'" <agh1@nrc.gov>
Date: 4/18/05 9:11PM
Subject: RE: SUPPLEMENTAL DRAFT INFORMATION ON INSTRUMENT UNCERTAINTY

As discussed during last Friday's conference call the supplemental draft information on instrument uncertainty is attached.
Bryan

-----Original Message-----
From: Thomas Alexion [mailto:TWA@nrc.gov]
Sent: Monday, April 18, 2005 3:43 PM
To: MILLER, D BRYAN
Subject: SUPPLEMENTAL DRAFT INFORMATION ON INSTRUMENT UNCERTAINTY

Bryan,

(I haven't seen anything yet?)

Please e-mail the information to me, Kaly, and Allen Howe. Allen's e-mail is agh1@nrc.gov. (I'll be in a training class tomorrow.)

Tom

Mail Envelope Properties (42645AD6.FE1 : 10 : 20449)

Subject:	RE: SUPPLEMENTAL DRAFT INFORMATION ON INSTRUMENT UNCERTAINTY
Creation Date:	4/18/05 9:11PM
From:	"MILLER, D BRYAN" <dmill14@entergy.com>
Created By:	dmill14@entergy.com

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AGH1 (Allen Howe)
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MESSAGE	507	04/18/05 09:11PM
Draft for NRC 4-18-05.pdf	126305	
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Expiration Date:	None
Priority:	Standard
Reply Requested:	No
Return Notification:	None
Concealed Subject:	No
Security:	Standard

[Insert Correspondence Number]

[Insert Date]

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

SUBJECT: License Amendment Request NPF-38-249-1
Extended Power Uprate (Amendment 199) License Condition Regarding Instrument Uncertainty
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

REFERENCES: 1. NRC letter to Mr. Joseph E. Venable dated April 15, 2005, "Waterford Steam Electric Station, Unit 3 – Issuance of Amendment Re: Extended Power Uprate (TAC No. MC1355)"

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Operations, Inc. (Entergy) hereby requests that the license condition regarding instrument uncertainty that was imposed on the Waterford Steam Electric Station, Unit 3 (Waterford 3) license in Reference 1 be deemed complete and removed from the Waterford 3 license.

Reference 1 approved the Extended Power Uprate (EPU) for Waterford 3 and, as part of the approval, imposed the following license condition:

3. *As stated in the licensee's letter dated February 5, 2005, the licensee committed as follows: "Prior to exceeding 3441 MWt, Entergy will submit, for NRC review and approval, a description of how Entergy accounts for instrument uncertainty for each Technical Specification parameter impacted by the Waterford 3 Extended Power Uprate." Accordingly, subject to completion of this condition, the licensee shall not operate the Waterford 3 facility at a power level exceeding 3441 MWt.*

Descriptions of how Entergy accounts for instrument uncertainty for each Technical Specification parameter impacted by the Waterford 3 EPU are provided in Attachment 1.

The information has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and it has been determined that the removal of the license condition involves no significant hazards consideration. The bases for these determinations are included in the attached submittal.

Entergy requests approval of the proposed amendment by May 27, 2005 to support power ascension from the Spring 2005 refueling outage. Once approved, the amendment shall be implemented prior to exceeding 3441 MWt.

Waterford 3 can not exceed 3441 MWt and achieve the EPU power level of 3716 MWt following the Spring 2005 refueling outage until the license condition imposed in Reference 1 is deemed complete and removed from the license. The need for a license amendment for this purpose was not recognized by Entergy or the NRC staff until just prior to the issuance of the EPU license. Therefore, to avoid a derating of Waterford 3 following restart from the Spring 2005 refueling outage, Entergy requests that this license amendment request be reviewed and approved on an exigent basis.

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 [insert date].

Sincerely,

J. E. Venable
Vice President, Operations
Waterford Steam Electric Station, Unit 3

XXX/ZZZ/yyy

Attachments:
1. Analysis of Proposed Technical Specification Change

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 70066-0751

U.S. Nuclear Regulatory Commission
Attn: Mr. Nageswaran Kalyanam MS O-7D1
Washington, DC 20555-0001

Wise, Carter, Child & Caraway
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P.O. Box 651
Jackson, MS 39205

Winston & Strawn
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Washington, DC 20006-3817

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

[Insert Correspondence Number]

Analysis of Proposed Technical Specification Change

1.0 DESCRIPTION

This letter is a request to amend Operating License(s) NPF-38 for Waterford Steam Electric Station, Unit 3 (Waterford 3), to remove the license condition regarding instrument uncertainty that was imposed on the Waterford 3 with the approval and issuance of the Extended Power Uprate (EPU) amendment. The removal of the license condition will allow Waterford 3 to exceed 3441 MWt and achieve the EPU power level of 3716 MWt.

2.0 PROPOSED CHANGE

Remove license condition regarding instrument uncertainty that was imposed on Waterford 3 with the approval and issuance of the EPU amendment.

3.0 BACKGROUND

The amendment approving the EPU for Waterford 3 imposed the following license condition:

3. *As stated in the licensee's letter dated February 5, 2005, the licensee committed as follows: "Prior to exceeding 3441 MWt, Entergy will submit, for NRC review and approval, a description of how Entergy accounts for instrument uncertainty for each Technical Specification parameter impacted by the Waterford 3 Extended Power Uprate." Accordingly, subject to completion of this condition, the licensee shall not operate the Waterford 3 facility at a power level exceeding 3441 MWt.*

4.0 TECHNICAL ANALYSIS

In accordance with the license condition, Entergy Operations, Inc. (Entergy) is documenting the treatment of instrument measurement uncertainty for parameters which were revised in association with EPU or pertinent to EPU analyses that fall within the following criteria:

* The parameter is a value which is measured using plant equipment. That is, the parameter is directly indicated to operators using installed plant instrumentation.

* The parameter is a value which is specified by a Limiting Condition for Operation (LCO) of the Waterford 3 Technical Specifications. Parameters listed in Technical Specification Tables which are called out by LCO's are considered within the scope of this effort. When an LCO refers to values specified in the Core Operating Limits Report (COLR), such values would also be considered within the scope of this effort.

This criteria considers parameters which are pertinent to power uprate analyses, even if the value of the parameter is unchanged for EPU. That is, the parameter is considered if of at least moderate importance for analyses pertinent to the parameter (e.g., analyses discussed in Bases of Technical Specifications(TS)) which had to be reperformed to support EPU. This criteria would capture parameters for which margins to acceptance criteria for analyses discussed in the Bases of applicable Technical Specifications have been impacted for EPU.

The parameter selection was discussed with the NRC staff during a conference call on 14 April 2005. The NRC staff concurred with the list of parameters provided by Waterford 3, with the proviso (agreed to by Entergy) that Containment Spray Riser Level (TS 4.6.2.1) also be included.

Values relating to applicability (e.g., MODES) of the Technical Specifications are considered to be generally out of the scope of the license condition. For example, Technical Specifications 3.2.1 through 3.2.4 for power distribution parameters (Azimuthal Power Tilt, Planar Radial Peaking, Linear Heat Rate, Departure from Nucleate Boiling Ration (DNBR) margin) are designated as applicable above 20% of Rated Thermal Power. The Entergy license condition scope will not include discussion of instrument uncertainties with respect to that 20% power criteria.

Entergy recognizes that safety analyses must account instrument uncertainty in all cases. Since the intent of many Technical Specifications is to provide assurance that the plant is within the assumptions of the accident analysis, it is appropriate that the instrument measurement uncertainties be accounted for in some manner. However, the level of rigor applied to documenting the instrument uncertainty and the associated accounting in the applicable analyses and procedures may vary based on the safety significance of the instrument function. Unlike limiting safety system setting (LSSS) values, there is no clear regulatory guidance describing specific methods that must be employed to address the instrument uncertainties associated with surveillances of Technical Specification parameters.

Waterford 3 has performed a categorization of Technical Specification parameters within the scope of the license condition. This categorization shown in the table below, also reflects discussions with the NRC staff on April 14 and 15, 2005. Parameters are classified as falling into one of four categories regarding treatment of instrumentation uncertainty:

Category	Description
A	Instrument Uncertainty is explicitly considered in analyses. There is an explicit offset between the Technical Specification value and the value assumed in the analyses pertinent to the Technical Specification.
B	Instrument Uncertainty is explicitly considered in plant surveillance requirements or alarm response procedures. There is an explicit offset between the LCO value in the Technical Specification and the value specified to be maintained in plant surveillance procedures.
C	The LCO value may also be the value assumed as initial conditions in safety analyses and the value specified to be maintained in plant surveillance procedures.
D	The Technical Specification value and the plant surveillance limit are the same and the parameter does not have an explicit analytical basis. The limited number of parameters in this category are based on engineering judgment.

More detail is provided below for the parameters of interest based on 14 and 15 April 2005 discussions with the NRC staff. None of the Technical Specification parameters impacted by the Waterford 3 Extended Power Uprate are classified in Category C. Although HICB-12 may allow Technical Specification parameters to be the same value as assumed in the safety analyses and specified in the plant surveillance procedures, due to the timing of the review process, Entergy has explicitly applied offsets for instrument uncertainty in the analysis or in

surveillance procedures for the Technical Specification parameters impacted by the Waterford 3 Extended Power Uprate.
Consistent with the approach to instrument uncertainty endorsed in HICB-12, Waterford 3 is applying a less rigorous (e.g., 1-sigma) measurement uncertainty to certain of the parameters listed as Category B items. Regulatory Guide 1.105 provides a methodology for achieving a 95/95 confidence factor for assuring that instrument setpoints are not adversely affected by uncertainty effects. There is no regulatory requirement that specifies the application or the amount of measurement uncertainty for TS LCO values. The LCO values of interest are initial condition values and do not serve as setpoints to actuate equipment to mitigate the impact of an accident. Thus, these LCO values are of much lower safety significance than instrument actuation setpoints, which serve, for example, to actuate the spray pumps in response to an event. Thus, from a risk and safety significance standpoint, justification exists consistent with HICB-12 to apply a smaller uncertainty. This has a safety benefit in terms of providing an increased operating range to plant operators and thus inherently lessening the burden on operations of a decreased operating range.

A discussion of parameters of interest follows the listing of pertinent parameters and their categorization.

Category	TS SECTION	Description	Tech. Spec. Value
A	1.24	Rated Thermal Power	3716 MW
A	2.2	Table 2.2-1: Linear Power Level-High	108% Rated Thermal Power
A	2.2	Table 2.2-1: Logarithmic Power Level-High	0.257% Rated Thermal Power
A	2.2	Table 2.2-1: Pressurizer Pressure - High	2350 psia
A	2.2	Table 2.2-1: Pressurizer Pressure - Low	1684 psia
A	2.2	Table 2.2-1: Containment Pressure - High	17.1 psia
A	2.2	Table 2.2-1: Steam Generator Pressure - Low	666 psia
A	2.2	Table 2.2-1: Steam Generator Level - Low	27.4% Wide Range
A	2.2	Table 2.2-1: Steam Generator Level - High	87.7% Wide Range
A	2.2	Table 2.2-1: Reactor Coolant Flow - Low	19.00 psid
B	3.1.1.4	Minimum T_{Cold} for Criticality	520°F
A	3.1.2.2	Boric Acid Makeup Tank (BAMT) Volume	TS Figures 3.1-1 and 3.1-2
A	3.1.2.8.a	Minimum BAMT Volume -- MODES 1,2,3,4	TS Figures 3.1-1 and 3.1-2
B	3.1.3.1	7" limit for Control Element Assembly (CEA) position with respect to rest of Group	7" (indicated position)
A *	3.1.3.1 ACTION b, c, d	CEA Misalignment criteria for ACTIONS	19" (indicated position)
A *	3.1.3.1 ACTION f	CEA Insertion criteria for ACTION f	145"
A *	3.1.3.5	145" Shutdown CEA Insertion Limit	145"
A *	3.1.3.6	CEA Regulating and Group P Insertion Limits	COLR Figure 5

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Category	TS SECTION	Description	Tech. Spec. Value
D	3.2.3 ACTIONS b.2 and b.3	Reduced Thermal Power requirements and Reduced Linear Power Level - High trip setpoints	50% RTP; 55% setpoint
D	3.2.3 ACTION b.3	≥95% Rated Thermal Power for verifying Azimuthal Tilt	95%
B	3.2.5	Reactor Coolant System (RCS) Flow Rate	148 Million lbm/hr
A	3.2.6	Tcold	≤549 deg F
A	3.2.6	Tcold	≥536 deg F
D	3.2.6 *	Tcold	≤559 deg F
A	3.2.8	Pressurizer Pressure	≥2125 psia and ≤2275 psia
A	3.3.1	Table 3.3-1 Applicability of Logarithmic Power Level-High trip (and NOTES)	10^{-4}% power
A	3.3.1	Table 3.3-1 Note (a) Logarithmic Power Level-High trip bypass reset	$3*10^{-5}$% power
A	3.3.2	Table 3.3-4: Containment Pressure - High	17.1 psia
A	3.3.2	Table 3.3-4: Pressurizer Pressure - Low	1684 psia
A	3.3.2	Table 3.3-4: Containment Pressure - High-High	17.7 psia
A	3.3.2	Table 3.3-4: Steam Generator Pressure - Low	666 psia
A	3.3.2	Table 3.3-4: Steam Generator delta P - High	123 psid
A	3.3.2	Table 3.3-4: Emergency Feedwater Control Valve Logic	36.3% Wide Range
A	3.3.3	Table 3.3-6: Control Room Intake Monitor setpoint	$\leq 5.45*10^{-6}$ µCi/cc
A	3.4.3.1.a	Pressurizer indicated level	≥26% and ≤62.5%
B	3.5.1.b	Safety Injection Tank (SIT) volume	>40% and <77.8%

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Category	TS SECTION	Description	Tech. Spec. Value
B	3.5.1.b	SIT volume mode applicability: 4 tanks operable below 1750 psia.	>39% and <77.8%
B	3.5.1 *	SIT volume mode applicability: 3 tanks operable below 1750 psia	$\geq$61% and <77.8%
A	3.5.1.d	SIT pressure	$\geq$600 psig and $\leq$670 psig
A	3.5.4.a	Reactor Water Storage Pool (RWSP) volume	>=83%
B	3.5.4.c	RWSP Maximum Temperature	<=100 deg F
A	3.5.4.c	RWSP Minimum Temperature	>=55 deg F
A	3.6.1.4	Containment Minimum Pressure	14.275 psia
B	3.6.1.4	Containment Maximum Pressure	27" w.g. (0.974 psig)
B	3.6.1.5	Containment Maximum Temperature	120F
B	3.6.1.5	Containment Minimum Temperature	90F
B *	4.6.2.1.a	Containment Spray Riser Level	149.5 ft MSL
B *	3.6.6.2	Annulus negative Pressure	> 5" WG
A	3.7.1.1	Table 3.7-2 allowed Reactor power with Main Steam Safety Valve's (MSSV's) Out-of-Service	85.3% and 66.7%
A	3.7.1.3	Condensate Storage Pool (CSP) volume	$\geq$ 92%
A	3.7.1.3	CSP minimum temp	$\geq$55°F
B	3.7.1.3	CSP maximum temperature	$\leq$ 100°F
D	3.7.1.7	Atmospheric Dump Valve (ADV) (automatic control)	> 70% RTP
A	3.7.4.A	Ultimate heat sink Wet Cooling Tower (WCT) basin level	>= 97%

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Category	TS SECTION	Description	Tech. Spec. Value
B *	3.7.4.B	Ultimate heat sink WCT Average Basin temp	≤89°F
A	3.7.4.C	Table 3.7-3: # Fans Required based on Wet Bulb and Dry Bulb temperatures.	Dry Bulb: 91°F & 98°F Wet Bulb: 75°F & 70°F
B	3.8.1.1	Diesel Fuel Oil Storage Tank Level	≥39,300 gal; >37,000 gal for 5 days
B	3.8.1.1	Diesel Fuel Oil Feed Tank Level	> 339 gallons
B	3.8.1.2	Diesel Fuel Oil Storage Tank Level	≥39,300 gal
B	3.8.1.2	Diesel Fuel Oil Feed Tank Level	> 339 gallons
B	3.9.10.1, 3.9.10.2, 3.9.11	23 feet water over irradiated fuel (over vessel flange when moving fuel)	23 ft

4.1 CEA Misalignment Criteria
Technical Specification 3.1.3.1 ACTIONs b, c and d:

These parameters are considered Category A, but merit discussion because the treatment of instrument uncertainty is explicitly built into the rod worth reactivity uncertainties which are then applied to indicated CEA position. Note also that the Waterford 3 treatment of this value is consistent with that of other Combustion Engineering (CE) NSSS plants.

Technical Specification 3.1.3.1 ACTION c addresses the condition of one CEA trippable but misaligned from any other CEA in its group by more than 19 inches; ACTION d addresses the condition of one or more CEA's trippable but misaligned from any other CEAs in its group by between the 3.1.3.1 LCO value of 7 inches (indicated position) and 19 inches. While these values are not being changed by EPU, this is considered a pertinent parameter for EPU due to potential changes in reactivity and rod worths for EPU core designs.

Note the 19 inches is defined as an Indicated position in ACTION b.

19 inches defines the difference between a large and small CEA misalignment. Per TS Bases, for small misalignments (less than 19 inches) of the CEA's, there is (1) a small effect on the time dependent long-term power distribution relative to those used in generating LCO and LSSS setpoints, (2) a small effect on the available Shutdown Margin, and (3) a small effect on the ejected CEA worth used in the safety analyses.

As discussed in TS Bases, the Core Protection Calculator System provides protection to the core in the event of a large misalignment of a CEA by applying appropriate penalty factors to the calculation to account for the misaligned CEA. With one or both Control Element Assembly Calculators (CEAC's) operable, this increased penalty factor is applied whenever the CEA has an outward deviation of approximately 9.5 inches or greater. Inward CEA position deviations are bounded by the CEA Misoperation (CEA Drop) analysis of FSAR Section 15.4.1.4; the analysis of this event for 3716 MWt EPU conditions was presented in Section 2.13.4.1.4 of the EPU report, letter W3F1-2003-0074. Figure 3 of the COLR, which does not require revision for EPU, provides the required power reduction after a CEA drop event. This 19 inch value was also the value specified in NUREG-0212, Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors, and in NUREG-1432, improved Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors.

The 7 inch value corresponds to the alarm setpoint for CEA position deviation, with an explicit treatment of instrument uncertainty.

Because rod worth uncertainties are determined as a function of indicated rod position, instrument uncertainty is accommodated within the analytical basis for the 19 inch parameter. Thus, it is not necessary to apply any explicit allowance for CEA position instrument uncertainty to this parameter in plant surveillance procedures since rod worth uncertainties are applied in the analysis.

4.2 CEA Insertion Limits
Technical Specification 3.1.3.1 ACTION f
Technical Specification 3.1.3.5
Technical Specification 3.1.3.6

Several Technical Specifications provide limits on group CEA positions or involve ACTIONS which are dependent on CEA positions. These parameters are considered Category A, but merit discussion because the treatment of instrument uncertainty is explicitly built into the rod worth reactivity uncertainties which are then applied to indicated CEA position. Note also that the Waterford 3 treatment of this value is consistent with that of other CE NSSS plants.

Technical Specification 3.1.3.1.f for a trippable but inoperable CEA within its alignment limits allows operation to continue if the rod is greater than or equal to 145 inches withdrawn or if it is within the Long Term Steady State Insertion Limits if in CEA group 6 or group P. The LCO for Technical Specification 3.1.3.5 requires that all shutdown CEA's be withdrawn to greater than or equal to 145 inches. Figures 4 and 5 of the COLR provide the insertion limits required by the LCO of Technical Specification 3.1.3.6, presenting limits on reactor power as a function of CEA group position in inches.

While none of these values, including COLR Figure 5, are being changed for EPU, these parameters are considered pertinent to EPU due to the potential changes in reactivity characteristics associated with EPU.

As discussed in Technical Specification Bases, the insertion limits of TS 3.1.3.5 and 3.1.3.6 ensure that (1) the minimum Shutdown Margin is maintained and (2) the potential effects of a CEA ejection accident are limited to acceptable limits. Small CEA misalignments would only have small effects on the time dependent long-term power distributions, on shutdown margin, and on CEA worths assumed for the CEA Ejection analyses.

Westinghouse procedures for calculating core physics inputs to safety analyses do not explicitly account for any uncertainty in measured group average CEA position. It is not considered necessary to explicitly account for such a factor since the Physics bias and uncertainty factors that are applied to the calculated worth of CEA's positioned at nominal insertion limits inherently account for the effect of CEA position uncertainty. These bias and uncertainty factors were based on the statistical analysis of differences between the calculated and measured CEA worth where the CEA worth measurement was obtained using the CEA Exchange technique. With this technique, the measured CEA worth is determined by relating the change in indicated position of the "reference" bank required to compensate the reactivity inserted by the "test" bank. Since no adjustments are made to account for uncertainties in the actual indicated CEA position, the tolerance limits obtained from the analysis of the raw measured and predicted worth will provide a conservative prediction of the actual worth at the indicated CEA position.

Note also that if the effects of CEA position uncertainty were explicitly included in the uncertainty analysis, the impact of the overall CEA scram worth uncertainty would be negligible. For an assumed lead bank position 3.7 inches beyond the assumed insertion limit, the associated reduction in CEA scram worth would be less than 0.5%. If this uncertainty component were statistically combined with the remainder of scram worth uncertainty of about 6.5%, the net uncertainty would increase by a negligible 0.02%.

4.3 Thermal Power limits associated with Azimuthal Tilt Actions Technical Specification 3.2.3 ACTIONs b.2 and b.3

Technical Specification 3.2.3 ACTION b.2 requires that Thermal Power be reduced to less than 50% of Rated Thermal Power within 2 hours if Azimuthal Tilt exceeds the value specified in the COLR. The Linear Power Level - High trip setpoints are to be reduced to 55% of Rated Thermal Power in the next four hours. ACTION b.3 specifies that the power operation at greater than 50% may proceed provided the Azimuthal Power Tilt is verified acceptable at 95% or greater of Rated Thermal Power. These parameters are considered pertinent to EPU since Rated Thermal Power is being revised, although these percentage limits are unchanged.

These parameters are considered Category D.

The ACTION b.2 values for reduced power levels were chosen to be arbitrarily small. No explicit calculations are performed to support these values. These values are chosen to be small enough so that the plant would not be challenging any power operating limits if azimuthal tilt exceeded the Technical Specification limits. As discussed in the BASES of NUREG-1432, this provides an acceptable level of protection from increased power peaking due to potential xenon redistribution while maintaining a power level sufficiently high enough to dampen any resulting azimuthal xenon oscillations while maintaining sufficient margin to design limits. Similarly, the reduced Linear Power Level - High trip setpoint is considered sufficient to ensure the assumptions of accident analyses regarding power peaking are maintained.

The ACTION b.3 value of 95% is chosen to be close to full 100% Rated Thermal Power. This provision to allow stopping the high frequency (every hour) surveillances of azimuthal tilt provides an acceptable exit once measured azimuthal tilt has returned to an acceptable value.

These values have been judged to be sufficiently conservative as to accommodate allowances related to instrument uncertainties. Given this nature, it is not necessary to apply any explicit instrument uncertainty upon these values for power levels specified in the ACTION statements of Technical Specification 3.2.3.

4.4 T_{cold} Follow Reactor Power Cutback Technical Specification 3.2.6

Footnote * to Technical Specification 3.2.6 allows the upper limit on Tcold to increase to 559°F for up to 30 minutes following a reactor power cutback in which (1) regulating groups 5 and/or 6 are dropped or (2) regulating groups 5 and/or 6 are dropped and the remaining regulating groups are sequentially inserted.

This variable is considered Category D.

This value is being revised from 568°F to 559°F for EPU, in conjunction with the change to the Tcold LCO; the LCO is being revised from a range of 541°F to 558°F to a new range of 536°F to 549°F. It is noted, as documented in TS Bases, that a 3°F allowance for instrument uncertainty is applied to Tcold in FSAR Chapter 15 accident analyses, resulting in an analysis range of 533°F to 552°F. The 568°F value in Technical Specifications was arbitrarily chosen to be 10°F above the upper limit of the LCO, on the basis that it is reasonable to allow some deviation for a short period of time (30 minutes) to allow recovery and subsequent plant stabilization after the reactor power cutback.

This value has been judged to be sufficiently conservative as to accommodate allowances related to instrument uncertainties. Thus, it is not necessary or possible to apply any explicit allowance for instrument uncertainty to this value.

4.5 Containment Spray Riser Level Technical Specification 4.6.2.1.a

The NRC staff requested that this parameter be added to the scope of this license condition during a conference call on 14 April 2005. This parameter is considered Category B and merits further discussion because a less rigorous instrument measurement uncertainty will be applied in surveillance procedures for this parameter. This is considered appropriate and consistent with the guidance of HICB-12 due to the low safety significance of this parameter.

Technical Specification surveillance 4.6.2.1.a calls for maintaining a 149.5 ft MSL riser level in the containment spray riser piping. The post EPU surveillance requirement for this instrument is 186 ft indicated which corresponds to 154.5 ft MSL. The purpose of this requirement is to minimize the time before containment spray enters containment to mitigate the impact of containment pressurization transients, such as a Loss of Coolant Accident (LOCA) or a Main Steam Line Break. The acceptance limit for containment pressure is 44 psig.

Note the lowest centerline elevation of the containment spray header is 158' MSL, only 8.5' MSL above the Technical Specification on riser level. The lowest header is a 6 inch Schedule 40S pipe, with a 6.065 inch inside diameter. Thus, there is little operational margin above the Technical Specification requirement to accommodate instrument uncertainty, as a level of less than 8.5 feet above the Technical Specification requirement would result in spray flow into containment from the nozzles on the lowest of the risers. Also, significant operator burden is required to maintain the level within the required band; this burden is reduced, with associated benefits to nuclear safety, with a more flexible approach to instrument uncertainty for this parameter.

The calculated uncertainty for this parameter is less than 5 feet. An allowance of 5 feet for instrument uncertainty would result in less than a one second delay in the delivery of spray flow to the containment. A 1 second delay has no impact on the peak containment pressure due to LOCA, which occurs prior to spray flow into the containment air volume. Due to the large total energies involved, a 1 second delay in the start of spray would have negligible impact on the long term calculation of containment pressure, where the response over the first 24 hours is considered to demonstrate that containment pressure has been lowered to no more than half the peak pressure by 24 hours. A 1 second delay has an impact of only 0.09 psi on the worst case MSLB peak pressure of 41.88 psig. Given the conservatisms in the analysis and the margin to the 44 psig acceptance limit, the spray riser level instrument uncertainty is considered of very small safety significance. Thus, it is considered consistent with HICB-12 to apply a less rigorous instrument measurement uncertainty in the plant surveillance procedures to demonstrate compliance with Technical Specifications.

4.6 Annulus Negative Pressure / Shield Building integrity Technical Specification 3.6.6.2

Technical Specification 3.6.6.2, "Shield Building Integrity," requires the annulus region to be maintained at a negative pressure of at least 5 inch water gauge (w.g.) during normal operation (i.e., Modes 1, 2, 3, and 4). The Technical Specification limit equals the initial annulus pressure assumed in the post-LOCA annulus pressurization calculation. This parameter is not being changed as a result of EPU but is deemed to be pertinent to EPU since EPU radiological dose calculations are constructed on the basis of assumptions intended to represent or bound this value.

This parameter is considered Category B and merits further discussion because a a less rigorous instrument measurement uncertainty will be applied in surveillance procedures for this parameter. This is considered appropriate and consistent with the guidance of HICB-12 due to the low safety significance of this parameter.

The containment systems consist of the steel containment vessel surrounded by the Shield Building. The Shield Building provides biological shielding and controlled release of the annulus (region between the containment vessel and the shield wall) atmosphere under accident conditions, and environmental missile protection for the containment vessel and the Nuclear Steam Supply System.

The bases for this TS, as stated in TS Bases TS 3.6.6.2, is to ensure that the release of radioactive materials from the primary containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with operation of the Shield Building Ventilation System (SBVS), will limit the site boundary and control room radiation doses to within the limits of 10CFR50.67 during accident conditions.

The non-safety/non-seismic Annulus Negative Pressure System maintains a vacuum of at least 5 inch water gauge during normal operations to comply with the TS 3.6.6.2 Limiting Condition for Operation (LCO). Following a LOCA and receipt of a Safety Injection Actuation Signal (SIAS), the Annulus Negative Pressure System is deactivated and a transient condition exists in the shield building annulus until the SBVS is in full stable operation. (TS 3.6.6.1 requires the SBVS to be operable.)

In the post-LOCA annulus pressurization calculation, the initial annulus pressure is assumed to be -5 inch w.g. as protected by TS 3.6.6.2. This value is not algebraically adjusted in the calculation by the magnitude of the channel instrument uncertainty. The uncertainty for the annulus negative pressure instrument is 0.5 inch w.g. as documented in Waterford 3 uncertainty calculations. Indicated annulus negative pressure is typically maintained more negative than -7.7 inch w.g. by the automatic operation of the exhaust fans to ensure -5 inch w.g. is maintained in all parts of the annulus.

Calculations have shown that under the worst outside atmospheric conditions (e.g., temperature, humidity, etc.), and including instrument uncertainty in the pressure measurement, the initial annulus pressure may actually be less negative than the -5 inch w.g specified in the technical specification value. This could result in the annulus pressure becoming slightly positive for a short time early in the accident before the SBVS fans reduce annulus pressure.

To account for this condition, it is conservatively assumed that after the first 30 seconds post-accident, the annulus reaches a positive pressure and remains positive for 30 more seconds before the SBVS is able to maintain negative pressure in the annulus for the remainder of the event. This is consistent with BTP CSB 6-3, which requires that the total allowed containment leakage be assumed to be an unfiltered direct release to the environment when the shield building annulus pressure may be greater than -0.25 in. w.g. (i.e., TS 3.6.6.1 SBVS requirement) Thus, the dose contribution due to the assumed positive pressure period between 30 and 60 seconds after a LOCA is included in the total LOCA dose results. Note that only 40% of the total allowed containment leakage is assumed to leak into the shield building annulus region.

An informal calculation using the GOTHIC computer code has shown that operation of one SBVS fan restores negative pressure in the annulus in approximately 7.5 seconds This result demonstrates that the assumption of 100% unfiltered containment leakage to the environment for 30 seconds is a longer release time than it would take for the SBVS fans to restore a negative pressure.

Furthermore, under the Alternate Source Term methodology used for the Waterford 3 EPU radiological dose analyses (W3F1-2004-0053), the source term available for release is time dependent. Per RG 1.183, For the 30 seconds after the accident, only activity in the RCS water is available for release. The fuel-cladding gap gas activity is released starting at 30 seconds for a duration of one half hour (30 minutes). Due to these RG 1.183 timing assumptions, the relative activity in the containment is small at this time under AST assumptions compared to the containment activity at the end of the early in-vessel release phase, at 1.8 hours into the event. Thus, the impact of instrument uncertainty in the annulus negative pressure measurement on the calculated offsite and control room dose is very small.

Thus, there would be only a negligible-to-small impact on LOCA offsite and control room dose calculation if those calculations were to be performed assuming no initial vacuum in the shield building.

Therefore, the assumption of a 30 second unfiltered release results in a conservatively high offsite and control room radiological dose for this small contributor.

Because of the extremely small safety significance of this parameter, it is acceptable and consistent with HICB-12 to apply a less rigorous instrument measurement uncertainty to plant surveillance procedures to ensure Technical Specification compliance.

4.7 Power Level for OPERABILITY of ADV Automatic Actuation Technical Specification 3.7.1.7

This parameter is Category D.

New Technical Specification 3.7.1.7 is being added due to EPU to specify OPERABILITY required for the Atmospheric Dump Valves. This TS is being added since the EPU Small Break LOCA Emergency Core Cooling System (ECCS) analysis; the ADV's were previously credited only for cooldown to shutdown cooling entry conditions and for their containment isolation function. Thus, ADV operability, which had previously been addressed in the licensee controlled in Technical Specification 3.6.3 and the Technical Requirements Manual (TRM), will now be addressed in the new Technical Specification.

The small break LOCA analyses assume a maximum ADV setpoint of 1040 psia. This value is specified in the footnote to TS 3.7.1.7 and explicitly accounts for the instrument uncertainty offset from the nominal setpoint of 1007 psia.

The footnote to the LCO also documents that the ADV automatic actuation channels are not required to be operable when the reactor has been at less than or equal to 70% Rated Thermal Power for greater than 6 hours (following long-term operation at EPU Rated Thermal Power of 3716 MWt). The 70% is considered an arbitrary value to which uncertainty need not be applied. In support of this arbitrary value, analyses were conducted to demonstrate that the decay heat load associated with operation for 6 hours at 70% Rated Thermal Power is such that the ADV's need not be credited to demonstrate acceptable ECCS performance. The value of 70% is specified based on reasonable engineering judgment as a power level below which automatic actuation of the ADV's is not required. Note that the ADV's are not credited in the Waterford 3 Cycle 13 Small Break LOCA ECCS analyses, which leads to the conclusion that long-term operation at power levels of 3441 MWt (92.6% of EPU Rated Thermal Power) is acceptable without crediting ADV's in the SBLOCA analysis. The 6 hour time frame is consistent with ACTION a of new TS 3.7.1.7, which calls for reducing power to less than or equal to 70% of Rated Thermal Power within 6 hours if the automatic actuation channel for one ADV is inoperable and cannot be restored to operable status.

It is noted that there is generally no analytical basis for the ACTION times in Technical Specifications. The 6 hour action time here was chosen for consistency with Technical Specifications for similar functions, but that is an arbitrary time based upon shared engineering judgement which considers operating experience.

Margin exists in the decay heat analysis between that where ADV's are not required (e.g., long term operation at 3441 MWt) and the decay heat corresponding to operation at 70% Rated Thermal Power for 6 hours or less. A strict analytical approach would result in a curve of increasing Reactor Thermal Power as a function of time, that is, the reactor power could be slowly increased up to approximately 92.6% in order for this decay heat logic to be maintained. In consideration of this margin and the fact that the decay heat load associated with 70% power operation will decrease with longer times, it is not considered necessary to apply any explicit offset to account for power measurement uncertainty to the 70% value specified in Technical Specifications.

4.8 Wet Cooling Tower Basin Temperature Technical Specification 3.7.4.B

Technical Specification 3.7.4.b requires the wet cooling tower (WCT) basin water to be less than or equal to 89°F as a limiting condition of operation (LCO) for the ultimate heat sink (UHS). This limiting condition of operation ensures that the UHS can dissipate the peak accident heat load assuming the worst case meteorological conditions as required by Regulatory Guide 1.27. This parameter is considered pertinent to EPU due to higher decay heat for EPU conditions.

This parameter is considered Category B and merits further discussion because a a less rigorous instrument measurement uncertainty will be applied in surveillance procedures for this parameter. This is considered appropriate and consistent with the guidance of HICB-12 due to the low safety significance of this parameter.

Following the implementation of EPU, calculations demonstrate that the WCT basin temperature is required to be 89.3°F or less for the UHS to dissipate the peak accident heat load at the worst case meteorological conditions. The current LCO does not need to be changed since the LCO bounds the analysis value, ensuring the safety function of the UHS following the implementation of EPU will be met.

Performance testing of the component cooling water heat exchanger ensures margin in WCT basin temperature will exist. Analysis uncertainty is also applied to ensure actual UHS heat loads will not exceed the design basis limits. Thus testing and analysis provide conservatisms to the WCT basin water temperature.

A brief UHS system flow path description during accident mode is provided to understand how the testing and analysis briefly discussed above demonstrate the UHS safety function is met. The UHS consists of two systems, component cooling water (CCW) and auxiliary component cooling water (ACCW) systems. The major components in the CCW system for heat removal are the CCW pump, dry cooling towers (DCT), and the CCW heat exchanger (CCWHx). The major components in the ACCW system for heat removal are the ACCW pump, CCW temperature control valve (TCV), and the WCT. The CCW is a closed loop system with heat removal first being performed by the DCT. The DCT contains 5 cells of cooling coils with each cell being cooled by 3 fans. The remaining accident heat load will then be dissipated by the CCWHx. CCW flow enters the CCWHx and is cooled by the ACCW system. The CCW TCV controls the ACCW flow required to maintain CCW at the desired outlet temperature to cool the plant auxiliaries and remove accident heat loads. The heat removed by the ACCW is then dissipated to the atmosphere by the WCT. Each WCT contains a basin and two cooling cells, each cell consisting of 4 fans. Additional system description details of the UHS are provided in FSAR Section 9.2.5.

The ACCW system supplies cooling water to the CCWHx directly from the WCT basin. Therefore, the specific requirement for the WCT basin temperature LCO ensures the CCWHx will maintain desired outlet temperature to cool the plant auxiliaries and remove accident heat loads. The CCWHx is tested to comply with the requirements of the Generic Letter (GL) 89-13 program. The purpose of the GL 89-13 testing is to demonstrate that heat exchangers will perform their design basis heat removal function following a design basis accident. For CCWHx testing, data is collected while a more typical heat duty is being dissipated by the CCWHx. All measurement instrument uncertainties are applied to the measured data collected during testing. The data collected, after applying uncertainties, is analyzed to determine the overall health of the CCWHx and ensure the projected accident CCW outlet temperature meets the test acceptance criterion of 1.0°F or more less than the analyzed limit. If the acceptance criterion is met, a margin of 2.6°F exists in WCT basin temperature with respect to the basis of the LCO. In other words, if the WCT basin temperature was 91.6°F (89°F + 2.6°F), the CCWHx would dissipate its required peak accident heat load and maintain the CCW outlet temperature at the analyzed limit. This margin bounds the instrument uncertainty by most accurate available indication for this parameter.

The latest testing on the CCWHx demonstrated the CCW outlet temperature could be maintained more than 3°F below the analyzed limit. The current CCWHx testing assumes pre-

EPU conditions which are more severe than the EPU heat removal requirements, as discussed in Section 2.5.5.4 of the Power Uprate Report in W3F1-2003-0074.

The UHS peak heat load analysis assumes a CCW temperature control valve (TCV) uncertainty of -3.0°F which would increase the assumed UHS heat load post-accident and therefore is conservative. Cooler CCW temperatures to the plant auxiliaries result in removing more heat from the plant due to improved heat removal efficiency. The increase of CCWHX heat duty is expected to be 7.3MBtu/hr, the WCT basin temperature can be as high as 92.2°F and maintain the CCW outlet temperature at the analyzed limit.

Thus, conservatisms in the analysis methodology and inherent in GL 89-13 heat exchanger testing lead to the conclusion that the 89°F value is sufficiently conservative as to accommodate allowances related to instrument uncertainties. The margins described above provides assurance that the UHS will fulfill its safety function and does provide a bases for not explicit applying instrument uncertainty for the Technical Specification temperature limit. However, to insure Technical Specification compliance, an explicit, but less rigorous, uncertainty will be applied, consistent with HICB-12, to plant surveillance procedures.

5.0 REGULATORY ANALYSIS

5.1 <u>Applicable Regulatory Requirements/Criteria</u>

Per 10CFR50.36(c)(2), Limiting Conditions for Operation (LCO's) are the lowest functional capability or performance levels of equipment required for safe operation of the facility. It is not necessarily required to include explicit offsets for instrument measurement uncertainty to provide this required functional capability. Neither does 10CFR50.36(c)(2) prescribe any specific approach for the treatment of instrument measurement uncertainty. Thus, consistent with other industry precedents to base Technical Specification values on indicated values or to tie analyses to nominal values, the Waterford 3 approach maintains compliance with 10CFR50.36.

The graded approach to instrument uncertainty is explicitly endorsed in Regulatory Guide (RG) 1.105, Revision 3, December 1999, "Setpoints for Safety-Related Instrumentation," and Branch Technical Position HICB-12, "Guidance on Establishing and Maintaining Instrument Setpoints" (June 1997).

RG 1.105 applies only to setpoints, which are considered of greater risk and safety significance than initial condition values. However, given that RG 1.105 endorses a graded approach to be applied to setpoints, this provides a precedent for also using a graded approach in addressing parameters which are initial condition values. Setpoints are of far greater safety significance since a setpoint results in actuation of mitigation equipment; the availability of mitigation equipment is of far greater impact on analyzed results of a transient than slight variations in the initial conditions assumed for the analysis. For example, there would be far greater impact on Chapter 15 Nuclear Steam Supply System (NSSS) analyses if the control element assemblies did not insert on a reactor trip signal than if there was a slight variation in the control rod worth from the assumed value. Similarly, small variations in temperature of safety injection fluid would have a much smaller impact than if the safety injection pumps did not respond to the event.

The different nature of the significance of an instrument setpoint, compared to an assumed initial condition value, is highlighted by Generic Design Criterion (GDC) 29: " *Protection against anticipated operational occurrences"*. The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences." GDC29 focuses on the performance of safety functions, that is, on the ability of mitigating systems to respond to events, rather than on the small variability in analysis results associated with slight variations in the value of assumed initial conditions for the safety analyses.

No regulatory requirements exist for the incorporation of instrument uncertainty in the operating envelope limits used as inputs to the safety analysis process, with the exception of initial power level. Regulatory Guide 1.49 establishes the requirement that safety analyses be performed for an initial power level that accounts for power measurement uncertainty. However, for plants other than Waterford 3, some approved analysis methodologies credit other uncertainties to support performing analyses without explicit consideration of power measurement uncertainty. Also, licensing basis analyses for low probability events that are considered "beyond design basis" are performed at the licensed power level, without uncertainty (e.g., Station Blackout, Anticipated Transient Without Scram (ATWS)).

The determination of the safety significance of instrument functions should consider all available information. This would include review of deterministic requirements, the impact on risk, and other available information. Consideration of the margin of safety associated with applicable parameters would be within this scope. This approach ensures reactor safety, complies with regulatory requirements, is based on sound engineering practices, and avoids unnecessary operating restrictions upon the plant. This allows attention to be focused in a manner to maximize the safety benefit.

Waterford 3 setpoints for Engineered Safety Feature Actuation System (ESFAS) are listed in Technical Specification Table 3.3-4. The LCO for TS Section 3.3.2 requires that the ESFAS trip setpoints be consistent with the values shown in Table 3.3-4. Reactor protective instrumentation setpoints are listed in Technical Specification Table 2.2-1. The Limiting Safety System Settings (LSSS) for Reactor trip setpoints, Technical Specification 2.2.1, requires that *reactor protective instrumentation setpoints be set consistent with the values of* Table 2.2-1. The Bases of TS 3/4.3.1 and 3/4.3.2 describe the basis for the explicit treatment of instrument uncertainty for ESFAS and Reactor Protection System (RPS) setpoints:

> "RPS/ESFAS Trip Setpoint values are determined by means of an explicit setpoint calculation analysis. A Total Loop Uncertainty (TLU) is calculated for each RPS/ESFAS instrument channel. The Trip Setpoint is then determined by adding or subtracting the TLU from the Analytical Limit (add TLU for decreasing process value; subtract TLU for increasing process value). ...."

The methodology used by Waterford 3 for RPS/ESFAS setpoints has been previously reviewed and approved by the NRC as documented in Amendment 113 issued September 5, 1995.

It is noted that, aside from RPS/ESFAS setpoints, flexibility exists for licensees to determine what methodology to use when instrument uncertainty is to be explicitly accounted for. Branch Technical Position HICB-12 Revision 4 dated June 1997 states that licensees may apply a less rigorous setpoint determination method for certain functional units and LCO's.

The accounting of instrument uncertainty for other than ESFAS or RPS setpoints is discussed in an NRC Task Interface Agreement Evaluation (TAC No. M95177) dated July 22, 1996. The NRC staff has previously recognized that, for instrumentation other than ESFAS or RPS, instrument uncertainty can be accounted for through plant safety analyses, Technical Specification limiting values, measured values, surveillance testing, or emergency procedures. The use of ISA standard S67.04 is not required and other methodologies can be used to account for instrument uncertainty.

Note some of the parameters in the Table above do not serve as instrument setpoints, but rather are assumed initial conditions for parameters in safety analyses. Since these parameters are not instrument setpoints, they are beyond the scope of RG 1.105. Consistent with this philosophy of ISA-S67.04, which is endorsed by HIGB-12 and RG 1.105, it is recognized that there is far greater safety and risk significance for parameters which serve as setpoints for accident mitigation equipment than for parameters which only serve as initial conditions for analyses of postulated events.

Entergy has determined that the proposed change does not require any exemptions or relief from regulatory requirements and does not affect conformance with any General Design Criterion (GDC) differently than described in the Updated Final Safety Analysis Report (UFSAR.)

5.2 <u>No Significant Hazards Consideration</u>

This letter is a request to amend Operating License(s) NPF-38 for Waterford Steam Electric Station, Unit 3 (Waterford 3) to remove the license condition regarding instrument uncertainty that was imposed on Waterford 3 with the approval and issuance of the Extended Power Uprate (EPU) amendment (i.e., Amendment 199). The license condition required that additional information regarding how instrument uncertainty is accounted for in Technical Specification parameters impacted by EPU be submitted for NRC staff review and approval. The required information was submitted with this license amendment request and approval of this request documents the completion of the NRC staff's review and approval as required by the license condition. The removal of the license condition will allow Waterford 3 to proceed above 3441 MWt and achieve the EPU power level of 3716 MWt as authorized in Amendment 199 to the Waterford 3 Operating License.

Entergy Operations, Inc. has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed amendment is to remove a license condition imposed on Waterford 3 with the issuance of Amendment 199 approving the EPU. The removal of the license condition will allow Waterford 3 to operate at the power level of 3716 MWt which has previously been evaluated and approved by the NRC staff as documented in Amendment 199 to the Waterford 3 Operating License.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed amendment is to remove a license condition imposed on Waterford 3 with the issuance of Amendment 199 approving the EPU. The removal of the license condition will allow Waterford 3 to operate at the power level of 3716 MWt which has previously been evaluated and approved by the NRC staff as documented in Amendment 199 to the Waterford 3 Operating License.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed amendment is to remove a license condition imposed on Waterford 3 with the issuance of Amendment 199 approving the EPU. The removal of the license condition will allow Waterford 3 to operate at the power level of 3716 MWt which has previously been evaluated and approved by the NRC staff as documented in Amendment 199 to the Waterford 3 Operating License.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Entergy concludes that the proposed amendment(s) present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.3 Environmental Considerations

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

7.1 Entergy letter to the NRC dated November 13, 2003, "License Amendment Request NPF-38-249, Extended Power Uprate" (W3F1-2003-0074)

7.2 Waterford 3 Final Safety Analysis Report
7.3 Waterford 3 Technical Specifications (through Amendment 199)
7.4 NUREG-0212, "Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors," Revision 3, December 1981
7.5 NUREG-1432, Improved Standard Technical Specifications Combustion Engineering Plants," Revision 3, June 2004
7.6 Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants"
7.7 Generic Letter (GL) 89-13, "Service Water System Problems Affecting Safety-Related Equipment
7.8 10CFR50.36, "Technical Specifications"
7.9 Regulatory Guide 1.105, "Setpoints for Safety-Related Instrumentation," Revision 3
7.10 Branch Technical Position HICB-12, "Guidance on Establishing and Maintaining Instrument Setpoints," June 1997
7.11 Generic Design Criterion 29, "Protection Against Anticipated Operational Occurrences"
7.12 NRC Task Interface Agreement Evaluation (TAC No. M95177) dated July 22, 1996
7.13 Regulatory Guide 1.49, "Power Levels at Nuclear Power Plants," Revision 1, December 1973