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Bradford Houston
Director, Nuclear Safety Assurance
Waterford 3

W3F1-2004-0042

May 13, 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 April 8, 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 civil/mechanical analysis. Entergy's response to these 14 questions is contained in the attachment 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 13, 2004.

Sincerely,

BLH/DBM/cbh

Attachment: Response to Request for Additional Information

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Attachment To

W3F1-2004-0042

Response to Request for Additional Information

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

Question 1:

On page 2.2-3, you indicated that the modal superposition method with a constant 3 percent damping was used for the branch line pipe break (BLPB) analyses. You also indicated that the response of entire reactor coolant system (RCS) to BLPBs was calculated using non-linear response time history analysis. Please define the non-linear response time history analysis and describe how modal superposition method is applied for the non-linear response time history analysis. Also, please confirm whether and how the use of 3 percent constant damping used for the EPU time history analysis is consistent with the design basis damping where the mass-stiffness damping of not more than 3 percent at significant modes of vibration was used.

Response 1:

Nonlinear time history analysis computes the response of the structure to time-dependent loads which are specified as excitation records. The non-linear parameters of the RCS BLPB analysis are the gaps and pre-loads applied at the RCS support boundary and in the reactor vessel internals (RVI) simplified/reduced model. They are listed in the current EPU Licensing Report (Attachment 5 to Entergy Letter W3F1-2003-0074) in Section 2.2.1.2. Excitation records for the BLPB analyses are in the form of forces. For RCS, the non-linear response time history analyses were performed to calculate the RCS response for all limiting Branch Line Pipe Breaks (BLPBs) due to EPU.

The modal superposition method in ANSYS begins by calculating the stiffness matrix of the RCS model in the starting configuration. The program monitors the simultaneous opening and closing of gaps at the preset gapped locations. It also monitors the changes in loads at preloaded interfaces and determines when a preloaded joint disengages and/or re-engages. These changes are determined at every integration step during time history solution. For each boundary configuration determined at any integration step, the program re-calculates stiffness and modal response to the set of time varying loading inputs.

Use of three percent (3%) damping across all modes was always considered appropriate based on the applicability of 3% for "equipment and large diameter piping for Safe Shutdown Earthquake (SSE)" per Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants". Per this guide, the modal damping values expressed as a percentage of critical damping should be used for viscous modal damping for all modes considered in an elastic spectral or time-history dynamic seismic analysis of the Seismic Category I structures or components. The modal damping values are used in the dynamic analyses associated with two different magnitudes of earthquakes, the Safe Shutdown Earthquake and the Operating Basis Earthquake or ($^1/_2$ the Safe Shutdown Earthquake). For equipment and large-diameter piping systems (pipe diameter greater than 12 in.) 3% damping is applicable for Safe Shutdown Earthquake (SSE) while 2% damping is applicable for Operating Basis Earthquake (OBE).

When the Waterford 3 RCS was originally analyzed for pipe breaks (main coolant loop breaks [MCLBs], pre-uprate), alpha-beta damping was the only method available. This only allowed the analyst to provide as much as 3% damping at the ends of the applicable frequency range, with all other modes damped at less than 3%. The current analysis for extended power uprate using the ANSYS modal superposition option allows 3% damping to be applied to all RCS modes of vibration, as was originally intended.

Question 2:

On page 2.2-4, you indicated that "[S]ince the simpler model has fewer loading points for the blowdown loads than the detailed model, some of the blowdown loadings are combined for the RCS analysis. The RVI [reactor vessel internal] model and sets of blowdown load time histories were used for the RCS primary side BLPB analyses." It appears that the simplified RVI model is not sufficient to contain the structural characteristics of the RVI. Please confirm whether and how the use of the simplified RVI model is adequate for the RVI and the RCS analysis subject to BLPB loading time histories. Also, please describe the applicable blowdown loadings and how these loadings were combined and applied to the RVI and RCS analyses to account for depressurization loading inside reactor vessel for the EPU loading conditions.

Response 2:

The approach of including a simplified RVI model within the reactor vessel (RV) model for the RCS pipe break analysis is the approach previously used and NRC-approved for all CE fleet plants. Note that the RCS pipe break analysis is not the RVI detailed loads analysis; the RVI loads analysis is performed using a much more detailed RVI-core model, with RV motion input from the RCS analysis as well as blowdown loading inputs.

The reduced RVI model is detailed enough to represent the mass, stiffness and non-linear connection of the support barrel (CSB), core shroud (CS), core, upper guide structures (UGS), lower support structure (LSS), and CSB snubbers. It is also detailed enough to represent hydrodynamic mass effects of the CSB-to-vessel and CSB-to-CS. Master degrees of freedom at mass nodes allow input of blowdown loadings on internals structures and the interior surface of the RV shell. Therefore, the simplified RVI model used in the RCS analysis enables the RCS response to a branch line pipe break to include the effects of the blowdown loads, RVI-core mass and CG, component stiffness, hydrodynamic mass and to conserve significant modes of vibration.

Besides response loads and displacements on RCS components and supports, the results of the RCS analysis therefore include RV motion imparted to the RVI at RVI-shell interface locations as input to the detailed RVI loads analysis. In generating the simplified RVI model, the characteristics of stiffness, mass, frequency, hydro-dynamic coupling and blowdown loads for RVI-core are equivalent to the detailed RVI-core model. Using this methodology, the RCS analysis generates the RV response motion which is applied back to the detailed RVI-core model for that loads analysis.

The only changes between the original RCS analysis and the EPU analysis were that (1) the original RCS with simplified RVI model was converted from STRUDL to ANSYS and (2) the input loadings were based on BLPBs at EPU conditions instead of on MCLBs at pre-uprate conditions.

The simplified model is intended to capture interaction effects between the RVI and RCS insofar as they would affect the input to the detailed RVI-core model. These interaction effects are not a major contributor to the input or to the structural response. There are sufficient masses in the simplified model to accurately represent the principal RVI-core model frequencies, but the use of a smaller number of masses would tend to over-predict the interaction effects.

Question 3:

On page 2.2-5, you indicated that the methodology is the same as previously used for RCS asymmetric loads analysis which was approved by the U.S. Nuclear Regulatory Commission (NRC or the staff) for main loop breaks. Please confirm whether the methodology includes applied asymmetric loads, computer codes and analytical models that are consistent with the current design basis analyses.

Response 3:

As noted in the Power Uprate Licensing Report, the methodology used for the RCS asymmetric loads analysis is the same as previously approved by the NRC. The differences in the details of the analyses are basically due to (1) the difference in input loadings due to BLPBs vs. MCLBs and (2) significantly increased computer capabilities. These details are discussed below (and in Chapter 3.9 of the Waterford 3 FSAR):

Main Coolant Loop Break (MCLB) Analysis for Pre-Uprate Conditions

The original asymmetric loads analysis of MCLBs was performed using a lumped parameter STRUDL model which included details of the reactor vessel and gapped and preloaded supports, major connected piping and components, and a simplified RVI-core model (as discussed in the response to Question #2). The pipe break thrust force, asymmetric sub compartment pressurization forces and asymmetric reactor internal hydraulic forces were applied to the model as simultaneous time history forcing functions. The pipe break thrust force was a suddenly applied pressure-area term applied at the postulated break location. The RV blowdown loads were spatially dependent loads that were calculated using the WATERHAMMER and CEFLASH-4 programs and applied to the RVI and to the inside surface of the RV. The STRUDL, WATERHAMMER and CEFLASH-4 computer codes were previously approved by the NRC for Waterford 3 (see FSAR Section 3.9.1.2).

For MCLBs postulated at the RV inlet and outlet nozzles, the break locations are inside the RV cavity. The subcompartment pressurization forces were therefore spatially dependent loads that were applied to the outside surface of the RV.

The time history responses of the model to MCLBs were determined in the STRUDL post-processor using the direct integration method with alpha-beta damping (see response to Question #1). The analysis generated RV responses which included component and support reactions and displacement and acceleration time histories.

Branch Line Pipe Break (BLPB) Analysis for EPU Conditions

Following elimination of MCLBs via leak-before-break (LBB) methodology, BLPBs in the largest tributary piping systems interfacing with the RCS (as listed in the EPU Licensing Report) replaced the MCLBs for consideration of dynamic effects of pipe break on the RCS. Due to the significant increase in computer capacity since the first Waterford 3 analyses were performed, the RCS model developed for the BLPB analyses for EPU is a full RCS lumped parameter ANSYS model, which includes details of the RV, the simplified RVI-core model, SGs, RCPs, MCL piping, and gapped, preloaded and non-gapped RCS supports. This full RCS model allows a more accurate analysis of the effects of postulated pipe breaks on RCS response. The simplified RVI-core model used for the EPU analysis was verified to be the ANSYS equivalent to the STRUDL model used in the original MCLB analyses.

The RV blowdown loads due to primary side BLPBs are calculated using the CEFLASH-4B program and are applied to the same model locations inside the RV as they were in the original RCS analysis. The ANSYS and CEFLASH-4B computer codes were previously approved by the NRC for Waterford 3 (see FSAR Section 3.9.1.2).

Like the original MCLB analysis, the thrust force, asymmetric sub compartment pressurization forces and asymmetric RVI hydraulic forces are applied as simultaneous time history forcing functions. The locations of the postulated BLPBs dictate where and on which component(s) the thrust force, applicable jet impingement loads and asymmetric sub compartment pressurization forces are applied. None of the BLPBs in the Waterford 3 plant are located within the RV cavity; most postulated BLPBs are located in the SG sub compartment, where the SGs and RCPs are located and therefore subjected to the asymmetric sub compartment pressurization forces.

The time history responses of the model to BLPBs are determined in ANSYS using the modal superposition method. Although ANSYS also has the direct integration capability, the modal superposition method is preferred because there is more analytical control on damping values. The analysis generates RCS responses which include component and support reactions and displacement and acceleration time histories.

Question 4:

On page 2.2-12, you indicated that a 10 percent increase in steam flow for the EPU condition will result in a 20 percent increase in the pressure differential across the dryer and results in a 50 percent higher stress in the dryer supports than currently reported in the analysis of record. Please provide an evaluation of the flow induced vibration of the steam dryer, dryer supports and flow-reflector with respect to fluid-elastic instability, acoustic loads and vortex shedding due to steam flow for the EPU.

Response 4:

The discussion below demonstrates that EPU steam flow will cause no unacceptable vibration effects on the steam dryers, dryer supports, or flow deflector plate.

A detailed flow-induced vibration analysis of the steam dryers and dryer supports has not been performed because there is direct design, analytical and operating experience from Palo Verde Unit 1 that shows the dryers and dryer supports will remain stable. The Palo Verde Unit 1 steam generators have been operating for nearly 20 years with higher steam flows through fewer dryers and have had no failures.

The Palo Verde steam generators were designed to transfer 3816 Mwt of power (3800 Mwt core power plus 16 Mwt from reactor coolant pump heat). Following the implementation of the EPU, the Waterford 3 steam generators will transfer 3716 Mwt of power. The following table provides a comparison of the salient parameters for both Palo Verde and Waterford.

Parameter	Palo Verde (Pre Uprate)	Waterford (Post Uprate)
Power Output (Mwt)	3816	3716
Steam Flow (Mlb/hr)	8.50	8.23
Steam Drum Diameter (inches)	232	253
Flow Area in Steam Drum (ft^2)	1174.3	1396.5
Number of Dryers	142	162
Dynamic Pressure (10^{-3} lbf/in^2)	0.197	0.167
Relative Fluid Dynamic Pressure	1.00	0.85

Since Palo Verde and Waterford 3 have the same steam dryer design, a direct comparison can be made regarding the effect of steam flow through the dryers. Palo Verde has higher steam flow through a smaller number of dryers. The forcing function that would cause flow-induced vibration will be 15% lower at Waterford after the EPU than at Palo Verde. Thus, it is concluded that the operating experience at Palo Verde provides adequate assurance that the EPU will not have an adverse effect on the steam dryers and that fluid-elastic instability will not occur.

Although the same comparison between Palo Verde and Waterford 3 can not be made for the flow deflector (Palo Verde has two main steam outlet nozzles), the overall stress on this component remains small compared to its allowable value. In addition, the natural frequency of the plate was shown in the analysis of record to be 34.7 Hz. Since this value is well above the frequency likely to cause resonance (approximately 20 Hz), there are no concerns that the increased steam flow will cause unstable vibrations of this component. In addition, even with increased steam flow following the EPU, the stress on the deflector plate is small (6.8 ksi) compared to its allowable value (29.1 ksi).

Question 5:

In reference to Section 2.2.2.1.4, "NSSS Component Evaluations, "please provide a summary of evaluation for pressurizer which is not evaluated in this section. The evaluation should include components in the lower end of pressurizer (such as the surge nozzle, lower head well and penetration, and support skirt) which are affected by the pressure and the hot leg temperature and the components in the upper end of the pressurizer (such as the spray nozzle, instrument nozzle, safety and relief nozzle, and upper head and shell) which are affected by the pressure and the cold leg temperature for operation at the uprated conditions. Also, please provide calculated stresses and cumulated fatigue usage factor (CUFs) for each component at the EPU and the current rated power conditions in comparison to the code allowable limits. Also, provide codes, and code editions used for the evaluation of pressurizer at the EPU condition. If different from the code of record, provide justification and reconcile the differences.

Response 5:

The design and operating temperatures of the pressurizer, and therefore its thermal movements, are not affected by power uprate. With respect to the pressurizer, the changes in the MCL piping T_{hot} and T_{cold} values due to power uprate do not change the thermal anchor motions (TAMs) at the surge line-to-pressurizer nozzle interface, as confirmed by the analysis described in Section 2.2.2.1.2 of the Power Uprate Licensing Report. Building thermal growths at the base of the pressurizer as well as the pressurizer design temperature (653°F), which affects thermal growth of the pressurizer, are unchanged for power uprate. Since the specified thermal transients (including surge line thermal stratification) remain applicable for EPU, as confirmed by the analysis described in Section 2.2.2.1.2 of the Licensing Report, the original thermal analyses remain bounding. Therefore, the original normal operation (NOP) pressurizer loads and movements remain valid for power uprate.

The postulated pipe breaks in the mechanical design of Waterford 3 and their effects on pressurizer support loads are not changed for extended power uprate or for LBB considerations. The original design basis of the pressurizer includes postulated BLPBs in the following pipelines:

- Pressurizer Spray Line
- Safety Valve Line A
- Safety Valve Line B
- Surge Line

The spray and safety valve line breaks occur at the interface with the pressurizer upper head. Loadings on the pressurizer from these breaks are not limiting with respect to the pressurizer design.

The limiting breaks for the pressurizer are surge line breaks, which were previously analyzed for pre-uprate conditions in the analysis of record (AOR). Analyses of surge line guillotine and slot breaks, which were postulated at and below the pressurizer surge nozzle, accounted for upward thrust, jet impingement and asymmetric pressure loadings on the pressurizer's lower head and support skirt. The AOR includes the 1.40 factor on the asymmetric pressure loadings which is only required at the preliminary design stage. Any small increase in asymmetric pressurization loads attributable to these surge line breaks under EPU conditions is more than offset by the 40% factor. For intermediate guillotine and slot breaks analyzed in the AOR at surge line locations further away from the pressurizer, the maximum moments applied to the pressurizer nozzle were based on the ultimate stress of the surge line pipe, thereby providing an upper limit on the moments that could be applied to the pressurizer nozzle due to surge line breaks. The stresses on the pressurizer, which are the source for the current rated power and EPU, are summarized in the table below. The indicated ASME Code Edition and Addenda are the same as those used in the analysis of record.

PRESSURIZER NOZZLES AND SUPPORT SKIRT
CRITICAL STRESSES AND CUFs

Component	Load Combination	Primary SI Class	Calculated SI (ksi)	Allowable SI (ksi)
Bottom Head Support Skirt	DWt + OBE + DP	Pm	25.81	26.7
	DWt + SSE + OP	Pm	23.44	41.4
	DWt + SSE + OP	P_L + Pb	26.10	62.1
	Faulted	P_L + Pb	30.33	70.55
	Primary + Secondary	P_L + Pb + Q	75.47	80.1
		CUF	*0.667*	*1.0*
Surge Nozzle	DWt + OBE + DP	Pm	21.20	26.7
	Primary + Secondary	P_L + Pb + Q	64.7	69.9
		CUF	*0.441*	*1.0*
	Faulted	P_L + Pb	83.64	84.0
Spray Nozzle	DWt + OBE + DP	Pm	16.27	26.7
	DWt + OBE + DP	P_L + Pb	24.65	34.95
	Primary + Secondary	P_L + Pb + Q	44.8	49.8
		CUF	*0.852*	*1.0*
	Faulted	Pm + Pb	36.48	84.0
Temperature Nozzle	DWt + OBE + DP	P_L + Pb	10.72	34.95
	Primary + Secondary	P_L + Pb + Q	29.05	69.9
		CUF	*0.033*	*1.0*
Upper Level Pressure Tap Nozzle	DWt + OBE + DP	P_L + Pb	8.91	34.95
	Primary + Secondary	P_L + Pb + Q	18.10	69.9
		CUF	*0.003*	*1.0*
Lower Level Nozzle	DWt + OBE + DP	P_L + Pb	8.91	34.95
	Primary + Secondary	P_L + Pb + Q	28.25	69.9
		CUF	*0.032*	*1.0*
Heater Tube Plug Weld	DP	Pm	7.40	16.6
	Primary + Secondary	P_L + Pb + Q	30.16	49.8
		CUF	*0.112*	*1.0*
Relief & Safety Valve Nozzle	Primary + Secondary	P_L + Pb + Q	33.83	80.1
		CUF	*0.004*	*1.0*
Heater Sleeve-to-Head Juncture	DWt + OBE + DP	P_L + Pb	31.92	34.95
	Primary + Secondary	P_L + Pb + Q	42.27	69.9
		CUF	*0.279*	*1.0*
Manway-Cover Plate Assembly	DP	Pm	20.95	26.7
	DWt + OBE + DP	P_L + Pb	26.59	40.1
	Primary + Secondary	P_L + Pb + Q	46.66	80.1
		CUF(ass'y)	*0.017*	*1.0*
		CUF(studs)	*0.023*	*1.0*

ASME BP&V Code, Section III, 1971 Edition, up to and including Summer 1971 Addendum.

Question 6:

In reference to Section 2.2.2.1.4.3.6, you concluded that the pre-uprate conditions bound the EPU conditions since the T_{hot} temperature for power uprate is lower than the temperature used in the analysis of record for pre-uprate, and the steam temperature remains the same. However, Table 1-2 shows that the hot leg temperature for EPU is higher than the current value, and the steam generator (SG) pressure (and the saturated steam temperature) is lower for EPU than the current rated value. Explain the apparent discrepancy.

Response 6:

The analysis of record that documented the structural integrity of the Waterford 3 steam generators assumed a hot leg temperature of 611 °F, a cold leg temperature of 553 °F and a saturated steam temperature (at 900 psia) of 532 °F. The corresponding steam generator conditions following implementation of the EPU are 601 °F, 543 °F, and 519.7 °F (saturated steam at 810.2 psia), respectively. The statement that the steam temperature remains the same is incorrect. It should have stated that the ΔT of the primary fluid remains the same.

As noted in Section 2.2.2.1.4.3.6, the parameters that affect the interaction between the tubes and the vertical grid supports are the coefficient of thermal expansion of the tube and supports (which does not change) and the average temperature between the fluid within the tube and the external steam temperature (i.e., the average metal temperature of the tube). In the analysis of record, the average metal temperature of the tube was (611 + 532)/2 = 571.5 °F on the hot side and (553 + 532)/2 = 542.5 °F on the cold side. Following the EPU the average metal temperature of the tube will be (601 + 519.7)/2 = 560.35 °F on the hot side and (543 + 519.7)/2 = 531.35 °F on the cold side. Since the average metal temperature will be lower following the EPU, the total tube bundle growth from ambient conditions to 100% power will be less. Therefore, the stresses calculated in the analysis of record will bound the EPU conditions.

Question 7:

In reference to Section 2.2.2.1.4.3, please provide a summary of the evaluation for the SG internals (baffle, feedwater sparger, steam dryer, flow reflector, tubes) and their supports with respect to the maximum stress and fatigue usage factor for the EPU conditions. Also, identify the Code and Code Edition for the evaluation of the proposed EPU and, if different from the Code of Record, please provide a justification.

Response 7:

Each component evaluation performed as part of the analysis of the secondary internals used the 1971 Edition of the ASME Code with Addenda through the Summer of 1971 as the Code of Record for both the original analysis of record and for the EPU conditions. In general, evaluation of the steam generator secondary internals addressed loads resulting from dead weight and seismic conditions as well as secondary flow and thermal conditions (where applicable). Since the secondary internals are not part of the ASME pressure boundary, most of these components satisfy the ASME Code requirements for not requiring an analysis for cyclic operation in NB-3222.4(d). For those components where fatigue was calculated (e.g., feedwater sparger) usage factors were below the ASME Code allowable value of 1.0.

The following table provides a summary of the maximum stress associated with each of the secondary internals as well as whether this condition occurred in the analysis of record (AOR) or during the evaluation of the EPU conditions.

MAXIMUM STRESS IN STEAM GENERATOR SECONDARY INTERNALS

Component	Maximum Stress	Allowable Stress	AOR or EPU
Baffle & Baffle Supports	24.1 ksi	56.3 ksi	AOR
Feedwater Sparger	7.3 ksi	53.7 ksi	AOR
Feedwater Supports	28.4 ksi	58.2 ksi	AOR
Can Deck	5.3 ksi	56.3 ksi	EPU
Egg Crates	19.9 ksi	45.9 ksi	EPU
Egg Crate Supports	9.6 ksi	22.5 ksi	EPU
Tie Rods	27.4 ksi	42.0 ksi	EPU
Dryer Supports	7.6 ksi	56.3 ksi	AOR
Flow Deflector	6.8 ksi	29.1 ksi	EPU
Bottom Blowdown Pipe	3.5 ksi	26.9 ksi	AOR

As can be seen from the above table, all stresses are well within the ASME Code allowable values with considerable margin.

Question 8:

In reference to Section 2.2.2.1.4.5.2, you indicated that inclusion of BLPB loads in the current evaluation necessitated a re-analysis of the Reactor Coolant Pump (RCP) safe end primary stresses for the faulted conditions. Provide a summary of evaluation for the crossover piping between RCP and SG, RCP nozzle safe ends at both the suction and discharge sides. The evaluation should include the calculated stresses, CUFs and code allowable limits.

Response 8:

Table 1 (in Section 2.2.2.1.4.5.1 of the Licensing Report) provides 'Primary Membrane plus Bending' stresses for RCS hot and cold leg piping for the Design and Faulted conditions. These stresses represent the enveloping stresses on main coolant loop piping and the RCP nozzle safe ends.

In CE fleet nomenclature, the piping between the RCP and SG is the 'suction leg' and the piping between the RCP and the RV is the 'discharge leg'. The cold leg piping comprises the suction and discharge legs. The table below expands Table 1 in order to provide the maximum 'Primary Membrane plus Bending' stresses on the hot leg, suction leg and discharge leg straight pipe and elbows separately.

MAIN COOLANT LOOP PIPING
PRIMARY MEMBRANE PLUS BENDING STRESSES & CUFs

Condition	Component	Stress (ksi)	Allowable (ksi)
Design	Hot Leg Pipe	10.05	27.6
	Hot Leg Elbow	18.63	27.6
	Discharge Leg Pipe	11.33	29.62
	Discharge Leg Elbow	24.06	29.62
	Suction Leg Pipe	13.16	29.62
	Suction Leg Elbow	23.56	29.62
Faulted	Hot Leg Pipe	12.34	55.90
	Hot Leg Elbow	24.30	55.90
	Discharge Leg Pipe	18.56	59.24
	Discharge Leg Elbow	40.96	59.24
	Suction Leg Pipe	21.64	59.24
	Suction Leg Elbow	40.87	59.24
CUF	Hot Leg Pipe	*0.382* *	*1.0*
	Hot Leg Elbow	*0.020*	*1.0*
	Discharge Leg Pipe	< *0.052*	*1.0*
	Discharge Leg Elbow	*0.052*	*1.0*
	Suction Leg Pipe	< *0.052*	*1.0*
	Suction Leg Elbow	< *0.052*	*1.0*

* due to MNSA holes in the hot leg straight pipe.

ASME BP&V Code, Section III, 1971 Edition, up to and including Winter 1971 Addendum.

Question 9:

In reference to Section 2.2.2.1.4.6.3, "Tube Vibration," the Waterford SG tubes were evaluated by calculating the most limiting fluid-elastic stability ratio and the maximum turbulent induced bending stresses on the limiting tube. Please provide a summary of evaluation regarding vortex induced vibration stresses on the limiting SG tubes for the EPU condition.

Response 9:

Vortex shedding manifests itself on a classic resonance situation when the Von-Karman vortex shedding frequency coincides with a tube span natural frequency. Theoretically when resonance exists, the magnification factor becomes infinite and even the smallest forcing function could produce vibrational failure. In reality, however, the forcing function must introduce more energy into the system than can be dissipated through damping or a stable level of vibration will be established. The vortex shedding frequency is described as follows:

fv = S (V/d)

where,

fv = Vortex shedding frequency
S = Strouhal No. (0.2 to 0.7)
V = Fluid gap velocity
d = Diameter of tube

However, results of the tube vibration testing revealed that tube bundles with tightly packed tube arrays were not susceptible to vortex shedding resonance. This also applied to the rotated square pitch pattern in the fluid exit region of this type of tube bundle. Thus, there is no need to evaluate vortex shedding in Waterford 3.

Question 10:

At Waterford 3, the limiting branch line pipe breaks replace the main coolant loop breaks (MCLBs) in the mechanical design basis, following elimination of MCLBs based on NRC approved Leak-Before-Break (LBB) technology. For the EPU evaluation, the analytical model consists of the entire RCS including RV, SG, RCP and their supports as well as control element drive mechanism (CEDM). Please provide calculated maximum stresses and CUFs for each of the major component supports including the pressurizer supports for the EPU and the current rated condition in comparison against the code allowable limits. Also, provide the code and code editions used for the EPU evaluation, if different from the code of record, provide a justification.

Response 10:

The calculated CEDM nozzle stresses vs. code allowable stresses are given in Table 2.2-2 of the Licensing Report. The CEDM nozzles were evaluated to the ASME BP&V Code, Section III, 1971 Edition, up to and including Summer 1971 Addendum. Stresses for the pressurizer and its support skirt are given in the response to Question #5. Stresses on supports for the other NSSS components are provided in the following tables. The indicated ASME Code Edition and Addenda for each component are the same as those used in the analysis of record.

RCP SUPPORT COLUMNS

Component	Upset Stress	Allowable	Faulted Stress	Allowable
Shearing				
Vertical Column	5.45	47.1	16.8	56.7
Upper Horizontal Link	2.36	39.1	6.1	48.3
Lower Horizontal Link	4.88	39.1	7.7	48.3
Compressive				
Vertical Column	4.07	28.2	12.0	36.1
Upper Horizontal Link	1.99	54.5	5.2	54.5
Lower Horizontal Link	0.27	54.5	5.2	54.5
Tensile				
Vertical Column	3.88	52.9	12.0	94.5
Upper Horizontal Link	1.99	43.9	5.2	80.5
Lower Horizontal Link	3.29	43.9	5.2	80.5

Units: ksi
ASME B&PV Code Section III, 1974 Edition through the 1975 Winter Addenda.
For compressive stress, the lower and upper upset condition stress limits were used for the faulted stress limits.

RCP SUPPORT CLEVISES

Component	Upset Stress	Allowable	Faulted Stress	Allowable
Shearing				
Vertical Column	1.51	32	4.44	46.2
Upper Horizontal Link	3.66	32	9.5	46.2
Lower Horizontal Link	5.19	32	8.2	46.2
Compressive				
Vertical Column	3.10	45.9	9.1	45.9
Upper Horizontal Link	2.52	45.9	5.4	45.9
Lower Horizontal Link	3.72	45.9	5.9	45.9
Tensile				
Vertical Column	3.90	36	11.45	57.8
Upper Horizontal Link	4.01	36	11.5	57.8
Lower Horizontal Link	5.64	36	8.9	57.8
Bending				
Vertical Column	1.7	60	5.0	96.3
Upper Horizontal Link	6.89	60	17.9	96.3
Lower Horizontal Link	10.2	60	16.1	96.3

Units: ksi
ASME B&PV Code Section III, 1974 Edition through the 1975 Winter Addenda.

RCP CLEVIS BASE PLATE

Component	Upset Condition		Faulted Condition	
	Bending Stress	Allowable	Bending Stress	Allowable
Vertical Column	0.667	60.0	2.0	96.3
Upper Horizontal Link	3.69	60.0	9.6	96.3
Lower Horizontal Link	5.44	60.0	8.6	96.3

Units: ksi
ASME B&PV Code Section III, 1974 Edition through the 1975 Winter Addenda.

RCP PINS

Component	Upset Stress	Allowable	Faulted Stress	Allowable
Bearing				
Vertical Column	9.56	132.3	N/A	N/A
Upper Horizontal Link	17.6	132.3	N/A	N/A
Lower Horizontal Link	10.9	132.3	N/A	N/A
Bending				
Vertical Column	6.73	110.3	19.8	144.4
Upper Horizontal Link	3.77	110.3	9.8	144.4
Lower Horizontal Link	8.84	110.3	14.0	144.4
Shearing				
Vertical Column	5.20	58.8	15.3	69.3
Upper Horizontal Link	3.51	58.8	9.1	69.3
Lower Horizontal Link	6.16	58.8	9.7	69.3

Units: ksi
ASME B&PV Code Section III, 1974 Edition through the 1975 Winter Addenda.

REACTOR VESSEL NOZZLES & SUPPORT FEET

Component	Load Combination	Primary SI Class	Calculated SI (ksi)	Allowable SI (ksi)
Inlet Nozzle	DWt+Th+OBE+DP	Pm	7.95	17.9
	DWt+Th+OBE+DP (in nozzle)	P_L + Pb	18.54	26.9
	DWt+Th+OBE+DP (at RV wall)	P_L + Pb	34.47	40.1
	Faulted Condition	Pm	34.20	56.0
	Faulted Condition	P_L	54.58	84.0
	Primary + Secondary	P_L + Pb + Q *CUF*	59.2 *0.19*	80.1 *1.0*
Outlet Nozzle	DWt+Th+OBE+DP	Pm	9.07	17.9
	DWt+Th+OBE+DP (in nozzle)	P_L + Pb	16.27	26.9
	DWt+Th+OBE+DP (at RV wall)	P_L + Pb	37.26	40.1
	Faulted Condition	Pm	35.5	56.0
	Faulted Condition	P_L	69.4	84.0
	Primary + Secondary	P_L + Pb + Q *CUF*	70.8 *0.40*	80.1 *1.0*
RV Supports	DWt+Th+OBE+DP	Pm + Pb	9.28	40.0
	Faulted Condition	Pm	29.17	51.0
	Faulted Condition	Pm + Pb	58.15	76.5
	Faulted Condition	Flange shear	30.16	30.6
	Primary + Secondary	P_L + Pb + Q *CUF*	11.79 *0.001*	80.1 *1.0*

Units: ksi
ASME BP&V Code, Section III, 1971 Edition, up to and including Summer 1971 Addendum.

REACTOR VESSEL BEARING SUPPORTS

	Support Plate	Bearing Plate	Slide	Socket	Expansion Plate
Level B Loading					
Bearing Calculated Allowable	 3.7 25.7	 3.7 58.0	 3.7 48.6	 3.7 49.8	 2.7 48.6
Shear Calculated Allowable	 0.9 10.3	 0.9 23.3	 0.9 13.5	 0.9 22.0	 0.0 13.5
Level D Loading					
Bearing Calculated Allowable	 19.1 30.8	 19.1 69.6	 19.1 48.6	 19.1 59.8	 37.0 48.6

Units: ksi
ASME B&PV Code Section III, 1974 Edition through the 1976 Winter Addenda.

SG SNUBBER LEVER ARM ASSEMBLY

	Link	Lever	Bracket	Snubber-to-Lever Attachment Bolts
Level D Loading				
Shear Calculated Allowable	 17.69 33.54	 32.21 58.70	 18.25 21.72	 7.14 33.54
Compressive Calculated Allowable	 12.15 36.75	n/a	n/a	n/a
Tensile Calculated Allowable	n/a	n/a	n/a	 38.96 56.01

Units: ksi
ASME B&PV Code Section III, 1977 Edition (no addenda)

SG SUPPORT SKIRT, SNUBBER SUPPORT LUGS & UPPER SHEAR KEY LUGS

Component	Load Combination	Primary SI Class	Calculated SI (ksi)	Allowable SI (ksi)
Support Skirt at Skirt Flange	DWt + Th + OBE	P_L + Pb	38.0	40.1
	DWt + Th + OBE	P shear	11.3	21.3
	Faulted Condition	P_L + Pb	58.5	76.6
Support Skirt at Head Interface	Faulted Condition	Pm	23.6	51.1
	Fatigue	*CUF*	*0.09*	*1.0*
Snubber Lug at Shell	DWt + Th + OBE	P_L	27.25	40.0
	DWt + Th + OBE	P_L + Q	40.8	80.0
	DWt + Th + SSE + DP	P_L	28.84	84.0
	Fatigue	*CUF*	*0.04*	*1.0*
Snubber Lug at Pin Hole	DWt + Th + SSE + DP	P shear	30.1	30.6
		P bearing	61.5	67.6
Upper Shear Key Lug at Shell	DWt + Th + OBE	P_L	12.9	40.0
	Faulted Condition	P_L	57.3	76.2
	Faulted Condition	P_L + Pb	27.15	84.0
	Fatigue	*CUF*	*0.04*	*1.0*

ASME BP&V Code, Section III, 1971 Edition, up to and including Summer 1971 Addendum.

SG SLIDING BASE

Component	Load Combination	Primary SI Class	Calculated SI (ksi)	Allowable SI (ksi)
Base Plate	DWt + Th + OBE	Pm + Pb	16.4	45.0
	Faulted Condition	Pb	33.4	63.0
	Faulted Condition	P shear	20.3	37.8
Anchor Bolt Washers	DWt + Th + DBE (Emergency)	P bearing	21.4	53.6
Bearings	DWt + Th + OBE	P bearing	5.10	31.14
Attachment Screws	DWt + Th + OBE	P bearing	15.85	46.7
Socket	Faulted Condition	P bearing	17.06	41.52
Expansion Plate	Faulted Condition	P bearing	41.81	63.0
Skirt-to-Base Stud	DWt + Th + OBE	P shear	18.72	29.9
	Faulted Condition	P tension	89.7 *	91.5
	Faulted Condition	P shear	29.9 *	54.9

* Calculated stresses are from the faulted combined load which includes MCLB. This combined load significantly envelops the faulted load combination due to BLPB.

ASME B&PV Code Section III, 1974 Edition through the 1975 Winter Addenda.

Question 11:

In reference to Section 2.2.2.2, you indicated that for the Waterford 3 EPU evaluation, all essential plant piping systems were evaluated to assess impact of various changes due to uprate and/or due to new BLPB LOCA (loss-of-coolant accident), thermal stratification, and new transient loading for the Chemical and Volume Control System (CVCS) charging and letdown lines. The EPU analysis results for piping systems were evaluated in accordance with the ASME Boiler and Pressure Vessel Code, Section III allowable. Provide the code editions which were used in the EPU evaluation. If different from the code of record, provide justification for applicability of the code editions.

Response 11:

Qualification of piping systems affected by EPU is consistent with the Code of Record for Waterford Unit 3. The Code of Record for each specific category of piping system is as specified below:

<u>Class I Piping</u>: ASME B&V Code, Section III, 1974 Edition, including addenda through Summer 1975.

<u>Class 2 and 3 Piping</u>: ASME B&V Code, Section III, 1971 Edition, including addenda through Winter 1972.

<u>Non-Safety/Non-Seismic Piping</u>: ANSI B31.1 Code, 1973 Edition.

Question 12:

On page 2.2-43, you indicated that "Piping qualification does not specifically evaluate for effects of vibration due to equipment or fluid flow. To ensure that changes resulting from EPU do not cause excessive vibration that could be detrimental to system performance, vibration monitoring will be performed following EPU to identify sources of vibrations and appropriate corrective actions will be taken to eliminate or minimize these vibrations." However, NB3622.3 requires that piping shall be designed so that vibration will be minimized. Please provide a summary of evaluation for flow effects on the main steam line vibration which will be higher for the EPU condition and describe your plan and schedule of the vibration monitoring program with regard to the power ascension, monitoring methods (installing accelerometers, using hand-held devices), strategic locations of monitoring, and acceptable criteria. Also, please confirm whether the vibration monitoring will be performed for both main steam and feedwater lines and branch lines, piping and components in accordance with the ASME Operations and Maintenance Code.

Response 12:

All essential piping systems in the plant are designed to withstand vibratory loading associated with a seismic event. Consequently, the design of the piping systems is inherently robust with respect to vibration. These plant systems were initially monitored for excessive vibration during initial power ascension and since then have been in operation for a number of years. Therefore, the piping systems have demonstrated acceptable vibration characteristics as required by the ASME code. EPU represents an incremental change in the system operating parameters, namely increases in flow in the main steam and feedwater lines, which could potentially alter the vibratory behavior of the piping system. Essential piping within these two systems is currently being monitored using pipe mounted instruments consisting of 8 accelerometers, two (2) on each of the Feedwater and Main Steam lines, placed at strategic locations to obtain baseline data for determining future changes in vibration characteristics. Following implementation of EPU, these locations will continue to be monitored to identify areas with potentially significant changes in vibratory behavior. Vibration data will be collected at the final power plateau (100%) as part of power ascension testing. Review of the data will be complete in time to support submittal of the Startup Test Report (within 90 days following startup). The vibration monitoring and evaluation of the measured data will be in accordance with ASME OM, Part 3, "Operations and Maintenance of Nuclear Power Plants". The acceptance criteria provided in ASME OM Part 3 shall be utilized. Compensatory and corrective actions will be taken, where required, to comply with the specified limits.

Question 13:

In reference to Section 2.5.5.3, you indicated that the heat loads on the Component Cooling Water (CCW) and Auxiliary Component Cooling Water (ACCW) systems are higher during normal shutdown. These higher heat loads increase the temperature of the CCWS return piping sections. Provide a summary of evaluation, including stresses and CUFs (if applicable) for CCWS piping, supports, and components such as heat exchangers, which are affected by the higher temperature in the proposed power uprate condition. In the evaluation, please include the Code allowable limits, Code including Code Edition used in the analysis.

Response 13:

The analysis of affected portions of the CCW system return piping downstream of the Shutdown Cooling heat exchangers is currently being evaluated for the expected temperature increase during normal plant shutdown. This is as committed to in the November 2003 licensing submittal in Section 2.5.5.3. Consistent with the current design, qualification of the affected portions of the piping system will comply with ASME B&V Code, Section III, 1971 Edition, including addenda through Winter 1972, which is the Code of Record for this piping system. The Shutdown Cooling heat exchangers, which are within the boundaries of the temperature increase, have been determined to be acceptable since the anticipated temperature increase is within the rated capacity of the heat exchangers. Increase in nozzle loading at the heat exchangers will be evaluated as a part of the piping qualification. In the unlikely event that these evaluations determine need for physical plant configuration changes, these changes will be implemented prior to operation of the plant under EPU conditions.

In addition to the aforementioned piping analyses, HVAC requirements in the rooms where the piping with the increased temperatures are located will be evaluated for acceptability. As in the piping analysis, in the unlikely event that these evaluations determine need for physical plant configuration changes, these changes will be implemented prior to operation of the plant under EPU conditions.

All analyses are expected to be complete by August 3, 2004.

Question 14:

Please discuss the functionality of safety-related mechanical components (i.e., all safety-related valves and pumps, including power-operated relief valves) affected by the power uprate to ensure that the performance specifications and technical specification requirements (e.g., flow rate, close and open times) will be met for the proposed power uprate. Also, please confirm that safety-related motor-operated valves (MOVs) in your Generic Letter (GL) 89-10 MOV program at Waterford 3 will be capable of performing their intended function(s) following the power uprate including such affected parameters as fluid flow, temperature, pressure and differential pressure, and ambient temperature conditions. Identify the mechanical components for which functionality at the uprated power level was not evaluated. Also, discuss effects of the proposed power uprate on the pressure locking and thermal binding of safety-related power-operated gate valves for GL 95-07 and on the evaluation of overpressurization of isolated piping segments for GL 96-06.

- Please discuss issues such as the design and testing for the atmospheric dump valves, which are now credited in the small break LOCA under EPU conditions, including the direct current voltage issue.
- Discuss the impact of higher flow rate under EPU conditions on the stroke time of applicable valves, such as the main steam isolation valves and feedwater isolation valves.
- Discuss any modifications of valves to support EPU operation, such as the moisture separator reheater safety valves.

- In Section 2.2.4 on page 2.2-52, the application states, "No changes to the programs related to GL 89-10, GL 96-05, or GL 95-07 are required." Section 2.2.4 also states that the safety-related air-operated, motor-operated, and hydraulic-operated valves were evaluated for any impact due to EPU conditions. Please discuss briefly, using a few example valves, the evaluations and the results obtained, including the effects of flow, ambient temperature, and voltage.
- Since vibration-induced performance problems are a primary issue for certain power uprates, please address this issue with respect to Waterford 3 EPU.

Response 14:

To clearly relate the responses to Question 14 to each of its several parts, the Question has been separated into eight parts as indicated below, each part followed by its corresponding response.

Question 14, Part 1:

Please discuss the functionality of safety-related mechanical components (i.e., all safety-related valves and pumps, including power-operated relief valves) affected by the power uprate to ensure that the performance specifications and technical specification requirements (e.g., flow rate, close and open times) will be met for the proposed power uprate.

Response 14, Part 1:

A review of all safety-related systems was performed as part of the EPU effort that resulted in the conclusion that the functionality of safety-related valves and pumps will not be affected or changed by the power uprate. All current actual performance characteristics and technical specification requirements will be maintained after implementation of the uprate. However, some applicable technical specification changes have been submitted for staff review. The discussion below is from the Power Uprate Submittal (Entergy Letter W3F1-2003-0074) and is given below for convenience.

Waterford-3 does not have power operated relief valves on the primary system.

<u>Main Steam Line Isolation Valves, Specification 4.7.1.5</u>

The full closure time for the main steam isolation valves is changed from the static test value of 4.0 seconds to the analysis value of 8.0 seconds which includes an assumed 1.0 second instrument response time. A closure time of 4.0 seconds, measured under static conditions, demonstrates closure under plant operating conditions within the 8.0 seconds assumed in the safety analysis. This change simply represents a preference for citing the closure time analysis value in the Technical Specifications. No change is being made to either the test closure interval or the closure interval assumed in the safety analysis.

<u>Main Feedwater Isolation Valves, Specification 4.7.1.6</u>

The full closure time for the main feedwater isolation valves is changed from 5.0 seconds to 6.0 seconds. Six seconds is the bounding analysis value that includes a 1.0 second instrument response time. The 5.0 second closure time did not include the 1.0 second instrument response time, which was separately accounted for in the safety analysis assumptions to arrive at a bounding analysis value closure time of 6.0 seconds. This change is made only to include the instrument response time in the analysis closure time cited in the Technical Specifications. Consistency is thus achieved with Specification 4.7.1.5 described above. No change has been made to the closure time of 6.0 seconds assumed in the safety analyses.

Atmospheric Dump Valves, Specification 3/4.7.1.7

This is a new Specification that addresses atmospheric dump valve (ADV) operability for the following:

- Plant cool down to shutdown cooling entry conditions with offsite power unavailable
- Small break LOCA (SBLOCA) mitigation above 70% of rated (EPU) thermal power.
- Containment isolation

The ADVs were previously credited only for cooldown to shutdown cooling entry conditions (primary safety function) and containment isolation (secondary safety function). ADV operability for cooldown is presently addressed in TRM requirement 3/4.7.1.7 while ADV operability for containment isolation is presently addressed in Technical Specification 3/4.6.3. There is currently no technical specification specifically addressing overall ADV operability.

For EPU, the ADVs are credited for SBLOCA mitigation at greater than 70% rated (EPU) thermal power. Therefore, a new Technical Specification ensuring ADV operability for this purpose is being proposed. Since a new ADV Specification is being added, it was deemed appropriate to address ADV operability for cooldown and containment isolation in the new Specification as well. Consequently, Technical Specification 3/4.7.1.7 is being created to address the cooldown, SBLOCA mitigation, and containment isolation functions of the ADVs.

Surveillance requirements and frequencies proposed for the ADV automatic actuation channels are consistent with those approved in Amendments 114 and 102 for South Texas Project Units 1 & 2 dated August 19, 1999 and are therefore considered to be appropriate. The surveillance requirement to cycle each ADV through a complete cycle is consistent with NUREG-1432, Rev. 2 and the current licensing basis as stated in the TRM. This requirement provides assurance that the ADVs can be used for cooldown and can be closed when needed for containment isolation. These surveillance requirements and frequencies are therefore considered to be appropriate for the ADVs.

The existing TRM requirement for ADVs, 3/4.7.1.7, is being deleted since it is now replaced by Technical Specification 3/4.7.1.7.

Question 14, Part 2:

Also, please confirm that safety-related motor-operated valves (MOVs) in your Generic Letter (GL) 89-10 MOV program at Waterford 3 will be capable of performing their intended function(s) following the power uprate including such affected parameters as fluid flow, temperature, pressure and differential pressure, and ambient temperature conditions. Identify the mechanical components for which functionality at the uprated power level was not evaluated.

Response 14, Part 2:

A review was performed of the Waterford 3 valve programs to determine how the Waterford 3 power uprate would affect valves in the Motor Operated/Air Operated/Hydraulic Operated valve programs (MOV/AOV/HOV). The basic methodology determined if the basis for determining the maximum expected differential pressure (MEDP) would change due to power uprate. Design basis review calculations for the program valves were obtained and reviewed for impact to the MEDP by expected power uprate effects. If the MEDP does not change or the change remains conservative relative to the assumed MEDP currently in the program, no further review was performed.

For the valves in the GL89-10 program, the existing MEDP calculation assumptions were found to be bounding for the power uprate (existing MEDPs the same or greater than expected for uprated conditions). Based on these evaluations, no physical changes or switch setting changes to MOVs are required for power uprate conditions.

Based upon the review of the valve programs, fluid flow rate was generally not an input into the MEDP calculations for globe and gate valves. However, it was an input to the MEDP calculations for butterfly valves and was reviewed accordingly. The flow rates used in the butterfly valve calculations were reviewed and are acceptable for power uprate conditions.

Process fluid temperature changes due to power uprate are considered insignificant relative to MOV capabilities and limits and were not reviewed for impact on Generic Letter 89-10 MOV program valves.

No increases in ambient temperature were identified that would impact Generic Letter 89-10 MOV program valves.

With regard to the question's interest in identification of mechanical components that were not evaluated: in general, unless a specific power uprate evaluation concluded that a change in performance is required of an existing system, it was concluded that there was no change in required performance expected of the system's components. In these cases, evaluations of individual components were not performed, with the exception of existing statements within the engineering report regarding safety related pumps and valves.

Question 14, Part 3:

Also, discuss effects of the proposed power uprate on the pressure locking and thermal binding of safety-related power-operated gate valves for GL 95-07 and on the evaluation of overpressurization of isolated piping segments for GL 96-06.

Response 14, Part 3:

<u>GL 95-07</u>

A review was performed to determine how the Waterford 3 power uprate might impact the existing Waterford 3 response to NRC Generic Letter 95-07, Pressure Locking and Thermal Binding of Safety Related Power Operated Gate Valves.

The required screening evaluation of all safety related power operated gate valves (48 valves) to identify those valves that are potentially susceptible to pressure locking or thermal binding was performed and is documented in a Waterford-3 report entitled "NRC Generic Letter 95-07

Screening Document", dated November 11, 1995. In that report, each valve was evaluated and categorized for the conditions of hydraulic locking, boiler effect, or thermal binding, as Not Susceptible, Non-Priority Susceptible, or Priority Susceptible. Valves that were categorized as Not Susceptible or Non-Priority were considered outside the scope of Generic Letter 95-07. The report concluded there were eight Priority Susceptible valves, which were susceptible to hydraulic locking. The Screening Document and the 180 day response to the NRC (Letter W3F1-96-0013, dated February 13, 1996) were reviewed for effects from power uprate and it was concluded that the results of the screening performed previously remained valid.

For valves that were initially considered susceptible (as considered in the pre-uprate evaluation, see above), physical modifications or further analysis had been performed to make them no longer susceptible. A review of these valves in support of the power uprate concluded that the modifications or analysis performed remained valid for uprate conditions. This is because the system conditions, the function of the valves and the physical configuration of the valves was not changing for the power uprate. Additionally, the ambient conditions surrounding the valves is not changing significantly for the power uprate.

In letter W3F1-99-0105, dated June 17, 1999, Entergy responded to an NRC request for additional information regarding why the Shutdown Cooling Suction Valves are not susceptible to pressure locking and thermal binding. This document was reviewed for the effects of power uprate. Since the containment temperature response is remaining essentially unchanged for the power uprate the overall conclusion that these valves are not susceptible to pressure locking and thermal binding remains valid.

It is therefore concluded that power uprate will have no adverse effect on the Waterford 3 response to NRC Generic Letter 95-07, Pressure Locking and Thermal Binding of Safety Related Power Operated Gate Valves.

<u>GL 96-06</u>

A review was performed to determine how the Waterford 3 power uprate might impact the existing Waterford 3 response to NRC Generic Letter 96-06, Assurance of Equipment Operability and Containment Integrity during Design Basis Accident Conditions. That NRC Generic Letter requested that licensees determine if piping systems that penetrate the containment are susceptible to thermal expansion of fluid so that over pressurization of piping could occur.

Letter W3F1-2001-0061 dated July 23, 2001, provided a summary of the Waterford 3 response regarding thermal expansion of fluid. That letter documented an evaluation/screening of all containment penetrations for susceptibility to thermally induced pressurization was conducted by Waterford 3 Design Engineering. Seventeen penetrations were found to be susceptible to thermally induced pressurization. Evaluations were initiated to address the subject penetrations and determine equipment operability along with the identification of corrective actions.

Six of the penetrations have had thermal relief valves installed to prevent overpressure. System configuration changes (valve lineups) were implemented for four containment penetrations to ensure the affected piping either remains in a drained condition during normal plant operations, or has a relief path for thermal expansion of trapped fluids. Waterford 3 elected not to install thermal relief valves in the remaining seven penetrations. The basis for this decision was:

- These seven penetrations are only vulnerable to overpressure in the relatively small timeframe that the plant is in heat up MODE 4.
- An evaluation of the pressure induced in these lines by a final water temperature of 260°F (post-LOCA containment temperature) shows that the penetrations may exceed normal yield stresses and experience plastic deformation, but would not catastrophically fail, therefore, the penetrations would retain their ability to perform their safety function and maintain containment integrity.
- Administrative procedural controls that will minimize the heat up of any trapped fluid in these seven containment penetrations.

To support power uprate conditions, a review of the Waterford 3 response to Generic Letter 96-06, the relevant letters to and from NRC, calculations, and Engineering Request packages were reviewed for potential impact from power uprate. The containment pressure/temperature analysis performed for power uprate shows that the peak temperatures after a LOCA remain below 260°F (current analysis temperature) and that the administrative procedural controls put in place for the seven penetrations discussed above have not been modified. Additionally, there have been no physical changes to the systems, system conditions or system operations such that these penetrations would be affected by the power uprate.

As far as the water hammer issues and the Containment Fan Coolers discussed in this Generic Letter are concerned, there are no physical changes, no configuration changes and no system operating conditions to the systems that would cause the system functions and operations to change from the pre-uprate conditions. Also, the containment response post-LOCA for power uprate is essentially unchanged from the pre-uprate conditions.

Therefore, operation at power uprate conditions will have no adverse effect on the Waterford 3 response to NRC Generic Letter 96-06, Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions.

Question 14, Part 4:

Please discuss issues such as the design and testing for the atmospheric dump valves, which are now credited in the small break LOCA under EPU conditions, including the direct current voltage issue.

Response 14, Part 4:

The ADVs are pilot-operated globe valves. Pilot-operated globe valves serve to reduce the required actuator thrust by balancing the upstream and downstream pressure before the main plug is lifted off the seat. The design requires that both the pilot valve and the main plug be considered when determining the minimum required thrust. In this particular instance, the pilot valve controls the thrust requirements in both the open and close directions. In the open direction, the differential pressure area of the pilot is greater than that of the main plug due to its balanced design. In the close direction, once the pilot valve is seated against the main plug, the fluid flowing over the main plug assists valve closure. Therefore, only the pilot valve needs to be considered.

The testing requirements have been discussed previously. In addition, these valves will be tested in accordance with the Waterford 3 In-Service Testing Program.

Due to the nature of the valve design, the direct current voltage issue is not applicable to these valves.

Question 14, Part 5:

Discuss the impact of higher flow rate under EPU conditions on the stroke time of applicable valves, such as the main steam isolation valves and feedwater isolation valves.

Response 14, Part 5:

Main Steam Isolation Valves

The calculation for determining the thrust requirements for the Main Steam Isolation Valves (gate valves) states that the flow rate after a steam line break will be approximately three times normal and is not a factor in the calculation. In general, process flow is not a significant factor when determining required thrust for gate valves since the flow will be greatly reduced by the time differential pressure begins to load the valve disk.

Feedwater Isolation Valves

The calculation for determining the thrust requirements for the Feedwater Isolation Valves does not use feedwater flow as an input in determining the MEDP and required thrust for these valves. During closure of these valves to isolate a feedwater line break, flow is expected to be much greater than normal. However, process flow is not a significant factor when determining required thrust for gate valves since the flow will be greatly reduced by the time differential pressure begins to load the valve disk.

Question 14, Part 6:

Discuss any modifications of valves to support EPU operation, such as the moisture separator reheater safety valves.

Response 14, Part 6:

Initial power uprate evaluations showed that the non-safety Moisture Separator Reheater (MSR) safety relief valves and the non-safety drain valves for the MSR drain tank and Feedwater Heater #6 would need to be replaced with greater capacity to accommodate uprate conditions. Evaluations are continuing as to what the final resolution of these valve replacements will be, if any.

Question 14, Part 7:

In Section 2.2.4 on page 2.2-52, the application states, "No changes to the programs related to GL 89-10, GL 96-05, or GL 95-07 are required." Section 2.2.4 also states that the safety-related air-operated, motor-operated, and hydraulic-operated valves were evaluated for any impact due to EPU conditions. Please discuss briefly, using a few example valves, the evaluations and the results obtained, including the effects of flow, ambient temperature, and voltage.

Response 14, Part 7

For an explanation of how evaluations for GL 95-07 were performed, see Response to Part 3 to this question.

The effects of flow and ambient temperature are discussed in the answer to Parts 2 and 5 above. It was not considered necessary to address the degraded voltage conditions for MOVs since EPU is not expected to cause degraded voltage conditions any different than those already analyzed for the GL 89-10 MOV Program valves. Specifically, the only changes to the Emergency Diesel Generator loads are the run times of the equipment powered by them.

The MOV, AOV and HOV evaluation methodology is discussed in the answer to Part 2 above. Typical examples of such evaluations are as follows. For the uprate conditions at Waterford 3, there are no expected voltage change issues or ambient temperature issues that would affect the valve open/close performance. Flow effects are limited to butterfly valves only. In those systems where there are butterfly valves, there are no changes to the process flows that would affect the valves' open/close performance.

Valves	Type	Evaluation Summary
Examples		
SI-225A/B, SI-226A/B, SI-227A/B, SI-228A/B, HPSI Header to RC Flow Control Valves	2" Globe (MOV)	MEDP OPEN of 1502 psid is based on upstream pressure of HPSI pump shutoff head plus static head from RWSP, and 0 psig downstream. NO IMPACT from Power Uprate. ---------- MEDP CLOSE of 1502 psid is based on upstream pressure of HPSI pump shutoff head plus static head from RWSP, and 0 psig downstream. NO IMPACT from Power Uprate.
ACC-110A/B, ACCW Pump Discharge Isolation Valves	16" Butterfly (MOV)	MEDP OPEN of 90 psid is based on upstream pressure of ACCW pump shutoff head of 204 ft plus elevation head from the WCT basin at overflow level, and downstream pressure of WCT basin at overflow level. Design flow used in calculation is 6500 gpm. NO IMPACT from Power Uprate. ---------- MEDP CLOSE of 90 psid is based on upstream pressure of ACCW pump shutoff head of 204 ft plus elevation head from the WCT basin at overflow level, and downstream pressure of WCT basin at overflow level. Design flow used in calculation is 6500 gpm. NO IMPACT from Power Uprate.
CVC-183, VCT Discharge Valve	4" Gate (MOV)	MEDP OPEN of 110.3 psid is based on upstream pressure of VCT pressure at relief valve setpoint plus backpressure of 87 psig, and downstream pressure of a perfect vacuum. NO IMPACT from Power Uprate. ---------- MEDP CLOSE of 110.3 psid is based on upstream pressure of pressure of VCT pressure at relief valve 2CH-R182A/B setpoint plus backpressure of 87 psig, and downstream pressure of a perfect

Valves	Type	Evaluation Summary
Examples		
		vacuum. NO IMPACT from Power Uprate.
EFW-223A/B, EFW HDR A/B to SG1/2 Backup Flow Control Valve EFW-224A/B, EFW HDR A/B to SG1/2 Primary Flow Control Valve	4" Globe (AOV)	MEDP OPEN of 1347.6 psid is based on suction from CSP, EFW pump (turbine driven) shutoff head of 3175 ft upstream, and 0 psig downstream. NO IMPACT from Power Uprate. -- MEDP CLOSE 1410 psid is based on suction from WCT basin with a suction head of 130 ft, EFW pump (turbine driven) shutoff head of 3175 ft upstream, and 0 psig downstream. NO IMPACT from Power Uprate.
MS-124A/B, Steam Generator 1/2 Main Isolation Valve	30" Gate (HOV)	MEDP OPEN of 100 psid is based on procedural controls that dP must be less than 100 psid prior to opening. Upstream pressure of 100 psig is from SG, with 0 psig downstream. NO IMPACT from Power Uprate. -- MEDP CLOSE of 1117.6 psid is based on the second lowest spring setpoint of the MS SRVs of 1085 psig plus 3% tolerance, and 0 psig downstream. NO IMPACT from Power Uprate.

Question 14, Part 8:

Since vibration-induced performance problems are a primary issue for certain power uprates, please address this issue with respect to Waterford 3 EPU.

Response 14, Part 8:

Industry Operating Experience (OE) was included in the evaluations of the systems. OEs that were reviewed for vibration issues were limited to large components such as steam dryers and feedwater heaters. The system evaluations performed for the power uprate included evaluations from the OEMs for MSRs, feedwater heaters, steam generators and the Main Condenser. All of the evaluations resulted in no issues with vibration with the exception of the Main Condenser. Evaluations are being performed to determine the extent of staking the condenser tubes to minimize tube vibration.