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Subject: Arkansas Nuclear One - Unit 2
Docket No. 50-368
License No. NPF-6
Response to Request for Additional Information from the Mechanical and Civil Engineering Branch Regarding the ANO-2 Power Uprate License Application

Gentlemen:

By application dated December 19, 2000, Entergy Operations, Inc. submitted an "Application for License Amendment to Increase Authorized Power Level." Pursuant to a request for additional information from the Nuclear Regulatory Commission (NRC) Mechanical and Civil Engineering Branch regarding the December 19, 2000, application, the attached information is submitted. The NRC request was received on June 21, 2001, and included thirteen questions. Entergy's verbal responses were discussed with the NRC staff during a telephone conference call on July 25, 2001.

Written responses are provided in two attachments since some of the information requested by the staff is the proprietary information of Westinghouse Electric Company, LLC. Non-proprietary responses to the staff's questions are contained in Attachment 1. Proprietary responses are contained in Attachment 2. A non-proprietary version of the responses contained in Attachment 2 will be submitted in the near future. Since Attachment 2 contains information proprietary to Westinghouse, it is accompanied by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the NRC and addresses the considerations listed in paragraph (b)(4) of Section 2.790 of *the Code of Federal Regulations*.

Accordingly, it is respectfully requested that the information proprietary to Westinghouse be withheld from public disclosure in accordance with 10CFR2.790.

Correspondence regarding the proprietary aspects of the information contained in Attachment 2 should be addressed to Mehran Golbabai, Project Manager, ANO-2 Power

AP01

Uprate, Westinghouse Electric Company, CE Nuclear Power LLC, 2000 Day Hill Road, Windsor, CT 06095.

Attachment 3 contains one regulatory commitment to complete the power uprate-related evaluations for Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," by September 30, 2001.

I declare under penalty of perjury that the foregoing is true and correct.

Very truly yours,

Dale E. James
Acting Director, Nuclear Safety Assurance

DEJ/dwb
Attachments

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Attachment 1 - Non-Proprietary Responses

Non-Proprietary Responses to the Mechanical and Civil Engineering Branch Request for Additional Information Regarding the ANO-2 Power Uprate

NRC Question 1

In reference to Section 5.3.3.2 of the application, provide the calculated maximum stresses and fatigue usage factors at the critical locations of the control element drive mechanisms for all operating conditions shown in Table 5-3 as a result of the power uprate. Also, provide the allowable Code limits for the critical components evaluated, and the Code and Code edition used for the evaluation. If different from the Code of record, justify and reconcile the differences.

ANO Response

The response to this question contains proprietary information. See Attachment 2.

NRC Question 2

Section 5.4 describes the mechanical and thermal analyses performed to determine the response of the reactor cooling system (RCS) main coolant loop and components, including the reactor vessel (RV), reactor coolant pumps (RCPs), replacement steam generators (RSGs), hot and cold leg piping, and component (RV, RCP, pressurizer and RSG) supports. The piping is discussed separately in Section 5.8. Provide the methodology, assumptions and loading combinations used for evaluating the RV, the pressurizer, the RCPs, the RSGs and their supports. Also provide the calculated maximum stresses and cumulative usage factors at critical locations of each component for the power uprate condition, including the allowable Code limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide a justification.

ANO Response

The response to this question contains proprietary information. See Attachment 2.

NRC Question 3

As a result of the RSGs and the power uprate, the feedwater flow and pressure in the feedwater system have to increase from those required for the RSGs at the current and uprate [sic] power levels. Discuss the potential for flow-induced vibration of the RSG tubes due to various mechanisms, including, in particular, the fluid-elastic instability in the RSGs at the current power level. Provide an evaluation of the flow-induced vibration of the tubes in the RSGs at the power uprate condition regarding the analysis methodology, damping value of the tubes and the computer code used in the analysis, results of the predicted vibration levels during the normal operating condition and the worst case transient condition, and the calculated fluid-elastic instability ratios. Explain whether or not the current analysis considers the potential for a possible degraded RSG condition.

ANO Response

The replacement steam generators were specified to be designed and analyzed for power uprate conditions including consideration of tube vibration and wear degradation. A general discussion of this is included in the response to Question 2.a of our letter dated August 7, 2001, "Response to Request for Additional Information from the Materials and Chemical Engineering Branch Regarding the ANO-2 Power Uprate License Application" (2CAN080101). To address the more detailed question above, the following additional information is provided.

Analyses and tests demonstrate that unacceptable tube degradation resulting from tube vibration is not expected for the replacement steam generators when operated at power uprate flows. Operating experience with steam generators having the same size tubes and similar flow conditions supports this conclusion.

Each replacement steam generator has eight tube support plates and five sets of anti-vibration bars with advanced design features. Alloy 690 thermally treated tube material and 405 stainless steel tube support material were selected to enhance the resistance to corrosion, mechanical wear, and fatigue. Anti-vibration bar widths are wider than in previous conventional designs to reduce wear potential. Accordingly, the increased steam flow rate following power uprate is not expected to result in a change in original design margin to instability or tube wear degradation at the anti-vibration bars. The cross flow in the lower straight leg portions of the tube bundle does not change appreciably since the increased feedwater flowrate is offset by a reduction in recirculation flow. Therefore, the potential for vibration/wear in this region is not significantly affected by uprate. A discussion of the replacement steam generator (RSG) design relative to tube vibration follows.

Potential sources of tube excitation are considered in the design, including primary fluid flow within the U-tubes, mechanically induced vibration, and secondary fluid flow on the outside of the U-tubes. The effects of primary fluid flow and mechanically induced vibration were evaluated and are acceptable. The main source of potential tube degradation due to vibration is the hydrodynamic excitation of the tubes by the secondary fluid. This area has been emphasized in both analyses and tests, including evaluation of steam generator operating experience. RSG thermal hydraulic modeling using ATHOS, a Westinghouse thermal hydraulic analysis code, determines the environmental conditions expected at the uprated power level. These environmental conditions are used as input to the RSG flow induced vibration evaluation.

Three potential tube vibration mechanisms related to hydrodynamic excitation of the tubes have been identified and evaluated. These include potential flow-induced vibrations resulting from vortex shedding, turbulence, and fluid-elastic vibration mechanisms.

Non-uniform, two-phase turbulent flow exists throughout most of the tube bundle. Therefore, vortex shedding is possible only for the outer few rows of the inlet region. Moderate tube response caused by vortex shedding is observed in some carefully controlled laboratory tests on idealized tube arrays. However, no evidence of tube response caused by vortex shedding is observed in steam generator scale model tests simulating the inlet region. Bounding calculations consistent with laboratory test parameters confirm that vibration amplitudes are acceptably small, even if the carefully controlled laboratory conditions were unexpectedly reproduced in the RSG.

Flow-induced vibrations due to flow turbulence are also small. Root mean square (RMS) amplitudes are consistent with levels measured in operating steam generators with benign tube wear experience. These vibrations cause stresses that are significantly below fatigue limits for the tubing material. Therefore, neither unacceptable tube wear nor fatigue degradation due to secondary flow turbulence is anticipated.

Fluid elastic tube vibration is potentially more severe than either vortex shedding or turbulence. Fluid-elastic tube vibration is a primary concern for anti vibration bar (AVB) wear. Testing performed by Westinghouse and field experience from previous designs have been utilized to develop analysis techniques to assure significant margin to instability is maintained. Linear dynamic analyses were performed covering a range of support configurations for various tubes using the finite element codes FLOVIB and FASTVIB. These are special purpose finite element codes that were written specifically for flow-induced vibration and fretting wear calculations for multi-span structural members. FLOVIB was written to incorporate the analytical approaches that were largely defined by the work of H. J. Connors at the Westinghouse Research Laboratories (later called Science and Technology Center). Three subprograms, SHAKE, GAMMA, and SUPER, comprise FLOVIB. Natural frequencies and mode shapes are determined in SHAKE. GAMMA uses SHAKE output and specified flow conditions in calculations of flow-induced vibration response of the structural member defined by beam elements. Peak and RMS values of selected GAMMA output parameters (displacements, stresses) are computed in SUPER.

Tube support spacing in the anti-vibration bars in the U-bend region provides tube response frequencies such that the required instability threshold limit is not exceeded for power uprate secondary fluid flow conditions. This approach provides large margins against initiation of fluid-elastic vibration for tubes, which are effectively supported by the tube support system. The largest stability ratio[1] for the Delta 109 steam generator tube bundle is 0.555 versus the conservatively specified 0.75 limit. The stability ratio is evaluated at power uprate loading conditions up to ten-percent tube plugging.

For the straight leg portion of the tubing, the worst case is where deposits are postulated to build between tubes and supports to the point where tube motion within the clearance is restricted or eliminated. For this case, frequency increases and damping reduces, so a separate evaluation was performed. When postulating limiting supports with reduced damping as a result of buildup, the largest stability ratio is 0.681.

Based on operating condition transient definitions, there are no Level A or B transients with higher than full-power steam flow, and normal operation analyses bound these transient

[1] Stability ratio defined as Fluidelastic Stability Ratio (FSR) = effective velocity/critical velocity

conditions. Level C and D transients are short duration events and are evaluated for bending stresses due to vibration in accordance with ASME B&PV Code requirements.

The Regulatory Guide 1.121 analysis for technical specification tube plugging limits addresses degraded tubes during normal and accident conditions at the uprated condition, including flow induced vibration loadings. A summary of this analysis has been previously submitted to the NRC in a letter dated July 19, 2000, "Regulatory Guide 1.121 Analysis for Arkansas Nuclear One Unit 2 Replacement Steam Generators" (2CAN070007). A discussion of the effects of power uprate on RSG degradation was included in the response to Question 2.b of our letter dated August 7, 2001, "Response to Request for Additional Information from the Materials and Chemical Engineering Branch Regarding the ANO-2 Power Uprate License Application" (2CAN080101).

As outlined, analyses and tests demonstrate that unacceptable tube degradation resulting from tube vibration is not expected for the RSGs when operated at power uprate flows. Operating experience with steam generators having the same size tubes and similar flow conditions supports this conclusion.

Table 3-1 summarizes the results.

Table 3-1 Summary Vibration Analysis Results for Expected Uprate Conditions Maximum for U-Bend Region and Straight Leg Region

Location	**Fluidelastic**			**Turbulence**		**Vortex**
	Maximum Stability Ratio, (Max FSR)	**Damping (%)**	**Frequency @ Max FSR (Hz)**	**Amplitude (RMS) Peak (10^{-3} in)**	**Stress (RMS) Peak (psi)**	**Amplitude (10^{-3} in)**
U-Bend, Tube R135C90	**0.555**	0.35	115.0	(0.44) 1.5	(71) 315	*
Peripheral Tubes-Straight Leg, R34C3-TSP Clean	0.423	1.5	41.4	(0.14)0.5	(37)164	<3.86
R1C4-TSP1-Plugged	**0.681**	0.9	84.6	(0.49) 1.7	(111) 494	-

*Shedding effects are bounded by turbulence correlations in the remainder of the bundle.

NRC Question 4

In regard to Section 5.2.2, you stated that for the holddown ring evaluation, rocking and sliding margins were calculated using the revised hydraulic input loads and moments, in combination with holddown ring loads derived from recent field ring deflection measurement data. Confirm whether and how the holddown ring is acceptable to provide adequate reactor vessel internal (RVI) hold down force and provide technical basis that the margin factors of 2 and 1.5 are considered acceptable as stated in Section 5.2.2. Also, in regard to Section 5.2.2, provide an assessment of flow-induced vibration of the RVI components due to the power uprate.

ANO Response

The response to this question contains proprietary information. See Attachment 2.

NRC Question 5

In reference to Section 5.7-1, you stated that following the application of leak-before-break (LBB), the remaining pipe breaks in the mechanical design basis of the RCS are all primary and secondary side branch line pipe breaks (BLPBs) interfacing with the RCS. Of these, the limiting breaks with respect to RCS structural considerations are breaks in the largest tributary pipes such as main steam line, feedwater line, surge line, safety injection line and shutdown cooling line. Clarify whether the thermal transient effects due to large-bore RCS pipe-break loss-of-coolant accidents (LOCAs) were considered in current licensing basis for the design of the ANO-2 RSGs. If not, explain why they were not considered (note that the approved LBB condition applies only to dynamic effects). Also, provide the stress analysis results for the primary side components of the RSGs including the RSG tubes to demonstrate the adequacy of the ANO-2 RSGs for the effects of thermal transients arising from postulated large-bore RCS pipe-break LOCAs during the power uprate.

ANO Response

Leak-before-break methodology was applied in the replacement steam generator design only for consideration of dynamic effects. The thermal effects during a loss of coolant accident (LOCA) were analyzed considering a large-bore reactor coolant system pipe-break. Since resulting thermal stresses are either secondary or peak stresses per Section III of the ASME B&PV Code, such stresses need not be considered in the Level D primary stress evaluation. In addition, since Level D events need not be included in the Section III fatigue assessment, and consistent with the requirements of Section III of the ASME B&PV Code, the thermal effects for a Level D (faulted) condition need only be considered in the Section III Appendix G nonductile fracture evaluation. A summary of the large break LOCA nonductile fracture evaluation results for the primary side components of the replacement steam generators is provided in Table 5-1. Consistent with the guidelines of Appendix G, the calculated critical flaw sizes are large (readily detectable) and thus are acceptable. It should be noted that the methods of evaluating nonductile behavior outlined in Appendix G are applicable only to ferritic materials. Non-ferrous materials such as alloy 690 used for the tubes exhibit ductile behavior even at relatively low operating temperatures, and thus brittle fracture of the tubes is not a concern.

Table 5-1 Summary of Large Break LOCA Nonductile Fracture Evaluation Results for Replacement Steam Generator Primary Side Components

Component	**Calculated Critical Flaw Size (inches)[1]**
Tubesheet (at channel head junction)	1.47
Primary Nozzle	> T / 4
Primary Manway	> T / 4

Notes:

[1]Reported flaw size is depth in the through wall direction. Consistent with Section III methods, the flaw length is six times the depth.

T = wall thickness

NRC Question 6

In reference to Section 5.7.2 [RCS Pipe Break Analysis Methodology], you stated that for the RCS with the RSGs, non-linear response time history analyses were performed to calculate the RCS response to the limiting BLPBs following the application of LBB technology. You also stated that a more detailed model of the RVI was included in the primary side pipe break model, because these pipe breaks cause RV blowdown loads. This RVI model included hydro mass and coupling terms, as well as additional nodes for RV blowdown input loadings. Confirm whether the analyses of the RV blowdown forcing functions and the non-linear structural responses due to the RSGs and the power uprate were performed by computer codes that were approved by the NRC or used in the analysis of record at ANO-2. Identify the computer codes that were used for the analyses of pipe breaks, seismic and transient events, that are different from those used in the original design basis analysis, and provide a justification that the new code was bench-marked for this application.

ANO Response

The response to this question contains proprietary information. See Attachment 2.

NRC Question 7

In reference to Section 5.7.2, you indicated that for the pipe break analysis of the RCS with RSGs, two three-dimensional ANSYS models of the entire RCS were developed from the RCS seismic model, one for secondary side breaks and one for primary side breaks. For the secondary side pipe break model, the representation of the RVI remained essentially the same as that for the seismic model, because secondary side breaks do not cause RV blowdown. A more detailed model of the RVI was included in the primary side pipe break model, because these pipe breaks cause RV blowdown loads. This RVI model included hydro-mass and coupling terms, as well as additional nodes for RV blowdown input loadings. The response of the entire RCS to pipe breaks was calculated using non-linear response time history analysis. The ANSYS computer code was used to perform the time history analyses due to BLPBs, using the modal superposition method and constant 3% modal damping. Clarify whether the ANSYS computer code was used to perform the non-linear time history analysis, using the modal superposition method. Describe the nonlinear parameters used in analysis. Also, provide a summary of analysis with a detailed model of the reactor internals to account for the depressurization blowdown loading in the BLPB analysis.

ANO Response

The response to this question contains proprietary information. See Attachment 2.

NRC Question 8

In reference to Section 5.8, provide, for the most critical RCS pipe systems evaluated, the calculated maximum stresses and fatigue usage factor, and code allowable limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide the necessary justification. Were the analytical computer codes used in the stress analysis different from those used in the original design-basis analysis? If so, identify the new codes and provide justification for using the new codes and state how the codes were qualified for such applications.

ANO Response

Section 5.8, "RCS Tributary Line Reconciliation Analysis" of the Power Uprate Licensing Report discussed the evaluation of changes resulting from the RSGs at power uprate conditions and reconciliation of the resultant loads against applicable code allowables. This discussion included safety injection, shutdown cooling, pressurizer spray, main steam, and main feedwater lines. The steam generator replacement necessitated the reconciliation analyses because the RSGs weigh more than the OSGs. When these analyses were performed for the RSGs, power uprate conditions were conservatively included. The impact of power uprate itself on the analyses was insignificant. In anticipation of a license extension for ANO-2, the piping was qualified for <u>60 years</u> (CUF are calculated based on 60-year life).

Analytical Computer Codes

As discussed in Section 5.8.2.2 of the PULR, the ME101 pipe stress analysis program was used to create mathematical models of the designated lines. Any piping stress reanalysis performed for the piping due to power uprate changes was performed using the Bechtel ME101 analysis software, which is a later version of the same analytical computer code (ME632) used for the original design basis analysis.

The ME101 program is an industry standard program that has been used on every facility designed by Bechtel. It performs the piping analysis in accordance with the ASME B&PV Section III or B31.1 Code formulae and rules. The Bechtel suite of programs containing ME101 is controlled under a Quality Assurance program that has been benchmarked and validated for changes that have occurred in the program since the analysis code version used during the initial design of ANO.

Code of Record

Main Steam and Main Feedwater

For the main steam and main feedwater lines inside containment, the Code and Code edition used in the analysis was the Code of record, which is ASME Boiler and Pressure Vessel (B&PV) Code Section III, NC/ND (Class 2/3), 1971 Edition through Summer 1971 Addenda.

Safety Injection, Shutdown Cooling, and Pressurizer Spray

Analyses for Class 1 piping for safety injection, shutdown cooling, and pressurizer spray were made per the 1980 ASME B&PV Code, Section III, NB-3600. Since the Code of record is the 1971 ASME B&PV Code through the Summer 1972 Addenda, a reconciliation was required. The following reconciliation applies to the stress analysis on the piping only.

The changes made to indices and stress equations from the Code of record to the 1980 Code are consistent with better understanding of piping stress. This understanding is derived from test and detailed finite element analysis. Using a more recent Code edition is not a problem since the analytical methods are not connected to evolving fabrication practices. The indices are established for standard piping components and weld types and changes in the specification of the components or welds are only allowed if it can be shown the indices are unaffected. Since the more recent Code edition has more joint types (definition of geometry for group or type of joint) than the older Code edition, some review is required to determine into which joint type in the more recent Code edition a joint fits. Since mixing codes is not recommended, all the piping reanalysis is done in the 1980 Code.

Allowables from the Code of record were used because the materials were tested and certified to meet the Code of record. 1980 Code allowables were permissible if the yield and ultimate are equal to those of the 1971 Code through Summer 1972 Addenda.

Generally, the ASME concurs with the use of more recent Code editions for stress analysis as long as the analyst is consistent and logical (ASME B&PV Section III 1986-NCA 1140). Therefore, the reanalysis was performed using the 1980 Code edition and the joint types were classified per the geometry limits in the 1980 Code. Allowables were taken either from the Code of record or from the 1980 Code edition where yield and ultimate equal the Code of record.

Calculated Maximum Stresses, Fatigue Usage Factors, Code Allowable Limits

This information is contained in the following tables. The bracketed numbers in the location descriptions refer to nodes used in the ME101 analysis.

As main steam and main feedwater are ASME B&PV Section III Class 2 piping, no fatigue usage factors are required for this piping.

Table 8-1
Safety Injection Line to B RCP
Class 1 Stress Summary Piping (2CCA-21 and 2CCA-51)

Description	Location of Max. Stress	Maximum Stress (psi)	Allowable (psi)	Ratio
Design < 1.5 Sm (Level A)	Connection to 1" vent line between core flood tank check valve and isolation valve [96]	21,441	27,450	0.78
Faulted <3.0 Sm (Level D)	Elbow nearest cold leg [15]	41,680	54,900	0.76
Primary plus Secondary Ranges (Equation 10)	Elbow nearest shield wall penetration [70]	57,701	54,900	1.05
Primary plus Secondary Ranges (Equation 12)	Elbow nearest shield wall penetration [70]	35,457	54,900	0.65
Primary plus Secondary Ranges (Equation 13)	Branch from 8" to 12" safety injection line [45]	27,422	54,900	0.50
Cumulative Usage Factor	Branch from 8" to 12" safety injection line [45]	NA	1.0	0.22

(Note: Bracketed numbers refer to nodes in the ME101 analysis)

Table 8-2
Safety Injection Line to A RCP
Class 1 Stress Summary Piping (2CCA-22 and 2CCA-52)

Description	Location of Max. Stress	Maximum Stress (psi)	Allowable (psi)	Ratio
Design < 1.5 Sm (Level A)	Connection to pressure transmitter [95]	15,888	27,450	0.58
Faulted <3.0 Sm (Level D)	Elbow nearest cold leg [10]	26,430	54,900	0.48
Primary plus Secondary Ranges (Equation 10)	Change from 6" to 8" line [240]	45,108	54,900	0.82
Primary plus Secondary Ranges (Equation 12)	Not required since Equation 10 is met.			
Primary plus Secondary Ranges (Equation 13)				
Cumulative Usage Factor	Connection to 3" HPSI header [235]	NA	1.0	0.13

(Note: Bracketed numbers refer to nodes in the ME101 analysis)

Table 8-3
Safety Injection Line to D RCP
Class 1 Stress Summary Piping (2CCA-23 and 2CCA-53)

Description	Location of Max. Stress	Maximum Stress (psi)	Allowable (psi)	Ratio
Design < 1.5 Sm (Level A)	Connection to 3" HPSI header [400]	21,046	27,450	0.77
Faulted <3.0 Sm (Level D)	Branch from 8" to 12" safety injection line [125]	31,870	54,900	0.58
Primary plus Secondary Ranges (Equation 10)	Connection to 3" HPSI header [400]	58,746	54,900	1.07
Primary plus Secondary Ranges (Equation 12)	U-bend between shield wall open penetration and connection to 8" line[100]	34,780	54,900	0.63
Primary plus Secondary Ranges (Equation 13)	Connection to 3" HPSI header [400]	34,205	54,900	0.62
Cumulative Usage Factor	Branch from 8" to 12" safety injection line [125]	NA	1.0	0.18

(Note: Bracketed numbers refer to nodes in the ME101 analysis)

Table 8-4
Safety Injection Line to C RCP
Class 1 Stress Summary (Piping 2CCA-24 and 2CCA-54)

Description	Location of Max. Stress	Maximum Stress (psi)	Allowable (psi)	Ratio
Design < 1.5 Sm (Level A)	Connection to pressure transmitter [201]	17,036	27,450	0.62
Faulted <3.0 Sm (Level D)	Connection to pressure transmitter [201]	23,910	54,900	0.44
Primary plus Secondary Ranges (Equation 10)	Branch from 8" to 12" safety injection line [35]	53,334	54,900	0.97
Primary plus Secondary Ranges (Equation 12)	Not required since Equation 10 is met.			
Primary plus Secondary Ranges (Equation 13)				
Cumulative Usage Factor	Branch from 8" to 12" safety injection line [35]	NA	1.0	0.20

(Note: Bracketed numbers refer to nodes in the ME101 analysis)

Table 8-5
Shutdown Cooling Line
Class 1 Stress Summary Piping (2CCA-25 and 2CCA-57)

Description	Location of Max. Stress	Maximum Stress (psi)	Allowable (psi)	Ratio
Design < 1.5 Sm (Level A)	Connection to pressure point between the two inside containment isolation valves [416]	23,042	27,450	0.84
Faulted <3.0 Sm (Level D)	Connection to pressure point between the two inside containment isolation valves [416]	29,320	54,900	0.53
Primary plus Secondary Ranges (Equation 10)	Elbow in line to pressure relief valve 2PSV 5085 [180]	58,508	54,900	1.07
Primary plus Secondary Ranges (Equation 12)	Elbow in line to pressure relief valve 2PSV 5085 [180]	46,144	54,900	0.84
Primary plus Secondary Ranges (Equation 13)	Connection to 3" line from HPSI header [12]	19,696	54,900	0.36
Cumulative Usage Factor	Connection to 3" line from HPSI header [12]	NA	1.0	0.64

(Note: Bracketed numbers refer to nodes in the ME101 analysis)

Table 8-6
Pressurizer Spray Line
Class 1 Stress Summary Piping (2CCA-13, -14, -15, -16)

Description	Location of Max. Stress (see table)	Maximum Stress (psi)	Allowable (psi)	Ratio
Design < 1.5 Sm (Level A)	[231]	13,920	27,450	0.51
Faulted <3.0 Sm (Level D)	[86]	36,650	54,900	0.67
Primary plus Secondary Ranges (Equation 10)	[108] [V1] [400] [10]	108,710 69,220 67,790 60,920	59,500 54,900 54,900 54,900	1.83 1.26 1.23 1.11
Thermal Expansion Moments (Equation 12)	[108] [V1] [400] [10]	24,310 6,800 43,910 48,840	59,500 54,900 54,900 54,900	0.41 0.12 0.80 0.89
Primary plus Secondary, without the Thermal Expansion Moments (Equation 13)	[108] [V1] [400] [10]	54,500 42,240 22,480 13,160	59,500 54,900 54,900 54,900	0.92 0.77 0.41 0.24
Cumulative Usage Factor	[V1],[V1A](1) [108] [140] [V20] [128] [106] [4]	NA		0.92 0.99(2) 0.96 0.89 0.88 0.86 0.82

Node	Description	Node	Description	Node	Description
[4]	Connection to main spray vent line	[108]	Elbow in auxiliary spray line	[400]	Connection to 1" line to reactor drain tank
[10]	Elbow upstream of main spray vent line	[128]	Bend in auxiliary spray line	V1	Connection on main spray vent line
[86]	Connection to 1" line to reactor drain tank	[140]	Elbow in auxiliary spray line	V1A	Elbow in main spray vent line
[106]	Elbow in auxiliary spray line	[231]	Reducer upstream of main spray valve 2CV-4656	V20	Reducer in auxiliary spray line to pressure point

(1) Node points V1 and V1A, are socket-welded connections on the ¾" Schedule 160 main Spray vent line which have been shown have acceptable CUF of 0.92. In addition, these locations have been shown to have acceptable stress levels when considered as Class 2 piping.

(2) Mostly due to conservative thermal stratification reanalysis.

Table 8-7
Main Steam Piping Inside Containment
(Piping 2EBB-1 and 2EBB-2)

Description	Location of Max. Stress	Maximum Stress (psi)	Allowable (psi)	% of Allow.
A Steam Generator				
Deadweight	Elbow at 47° H run [135B]	8,938	17,500	51.1
Maximum of Deadweight + OBE Seismic or Deadweight + Dynamic Steam Hammer Time History	Penetration [180]	18,973	21,000	90.3
Deadweight + (DBE Seismic + LOCA)[1]	Penetration [180]	24,265	42,000	57.8
Thermal Expansion + SAM-OBE Building Displacements	Elbow before penetration [175M]	13,592	26,250	51.8
B Steam Generator				
Deadweight	Elbow on 47° H run [135B]	9,115	17,500	52.1
Maximum of Deadweight + OBE Seismic or Deadweight + Dynamic Steam Hammer Time History	Containment penetration [170]	11,619	21,000	55.3
Deadweight + (DBE Seismic + LOCA)[1]	Steam generator connection [10]	20,835	42,000	49.6
Thermal Expansion + SAM-OBE Building Displacements	Containment penetration [170]	14,921	26,250	56.8

(Note: Bracketed numbers refer to nodes in the ME101 analysis)

(1) DBE Seismic and LOCA are combined by square root sum of the squares (SRSS)

Table 8-8
Main Feedwater/Emergency Feedwater Piping Inside Containment
(Piping 2-DBB-1 and 2-DBB-2)

Description	Location of Max. Stress	Maximum Stress (psi)	Allowable (psi)	% of Allow.
A Steam Generator				
Deadweight	Support upstream of check valve 2FW-5A [58]	7,578	17,500	43.3
Deadweight + OBE Seismic	Connection to pressure point line downstream of check valve 2EFW-9A [900]	12,961	18,000	72.0
Deadweight + (DBE Seismic + LOCA)[1]	Connection to steam generator [5]	26,414	42,000	62.9
Thermal Expansion + SAM-OBE Building Displacements	Connection to steam generator [5]	19,496	26,250	74.3
B Steam Generator				
Deadweight	Support downstream of check valve 2FW-5B [48]	7,459	17,500	42.6
Deadweight + OBE Seismic	Connection to 4" EFW line [32]	14,723	21,000	70.1
Deadweight + (DBE Seismic + LOCA)[1]	Shield wall penetration [5]	26,500	42,000	63.1
Thermal Expansion + SAM-OBE Building Displacements	Shield wall penetration [5]	19,514	26,250	74.3

(Note: Bracketed numbers refer to nodes in the ME101 analysis)

(1) DBE Seismic and LOCA are combined by square root sum of the squares (SRSS)

NRC Question 9

In reference to Section 2, you stated that the balance-of-plant (BOP) structures, systems and components have been evaluated for the impact of the 7.5 percent power uprate and in general found acceptable. Those requiring modifications due to power uprate consideration are provided in Table 2-2. Discuss the methodology and assumptions used for evaluating BOP piping, components, and pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers and anchorage for pipe supports. Were the analytical computer codes used in the evaluation different from those used in the original design-basis analysis? If so, identify the new codes and provide justification for using the new codes and state how the codes were qualified for such applications.

ANO Response

BOP piping, pipe supports, nozzles, penetrations, guides, and anchorage

The methodology used for evaluating BOP piping for the effects of power uprate was either the standard ANO engineering process for modifying systems or the ANO engineering methodology to evaluate systems not requiring physical modifications. If a piping configuration is modified to support the power uprate, the modified piping configuration is analyzed and qualified for the appropriate loading conditions as part of the normal modification process. The modification process qualification considered the post-power uprate parameters such as pressure and temperature and qualifies the piping stress, pipe supports, nozzles, penetrations, and anchors for the new configuration and loading.

There are very few modifications to piping as a result of power uprate, so the ANO engineering evaluation process was used for systems not being modified.

Piping systems that do not require modifications resulting from power uprate can still be affected by the uprate. Therefore, the following process was used to evaluate the impact of uprate on those systems. The primary design input parameters for piping analysis are the piping configuration and the loading. Since the physical configuration is not changing, the primary effect that power uprate might have on piping systems is driven by changes that affect pipe loads. The parameters that could affect loading if changed by power uprate are primarily pressure and temperature. For ANO-2, pressure and temperature parameters are determined and documented in "Pressure and Temperature" (PT) calculations. These PT calculations specify the maximum pressure and temperature values for each line class for the applicable plant operating modes (normal, upset, emergency, and faulted). Based on those maximum bounding design values, a temperature and pressure for each line class is established in the PT calculation. PT calculations affected by power uprate were revised to reflect the resulting changes. Revised PT values were evaluated for effect on piping stress, pipe support, nozzle and anchor qualification. The qualification of the piping systems for the revised PT values is documented in ANO calculations. In many of the systems, the changes in maximum pressure and temperature values due to power uprate were already bounded by design pressures and temperatures which therefore remain in effect. For those systems that did have a change that affected the design analysis of the piping system, the qualification of record for that piping was reviewed to identify the maximum stress and load values. Typically, a scaling factor was used to increase the stress or load calculated in the qualification of record by the ratio of the parameter increase. The new stress or load was then compared and documented to be within allowable limits. Typically, the increase in the input parameters was only a few psi or degrees, and the limits were not significantly challenged. The effects of the changes were specifically evaluated for the following aspects:

- thermal expansion stress
- pipe support loads
- nozzle qualifications
- flange and pipe fitting qualification
- pressure design (hoop stress) of piping systems
- creation of new high energy piping systems for HELB/MELB effects
- creation of new missile hazards from pressurized piping systems
- Flow Accelerated Corrosion acceptance criteria for minimum wall thickness limits
- piping thermal movement limits in fire barrier penetrations
- past flaw evaluations
- expansion joints
- dynamic loading due to fast valve closure transients.

The piping stress analysis does not use flow rate as a direct loading input. Changes in flow rate were considered from a structural perspective only with regard to the effect that changes in the flow rates would have on flow-induced vibration of the piping or on the dynamic loading due to fast valve closure transients. The response to Question 11 discusses the impact on flow-induced vibration.

Because the pressure and the mass flow rate will increase with power uprate, the main turbine stop valve fast closure transient analysis was updated, creating new dynamic forcing functions for the main steam headers and all hydraulically attached branch piping greater than four (4) inches. Bechtel evaluated the new forcing functions against the original dynamic analysis. The original analysis is bounding for the new forcing functions with the exception of the branch lines for the main steam supply to the main feed water pump driver turbines and the main steam supply to the second stage moisture separator-reheater tube bundles. A reanalysis of those four lines is being performed to qualify the piping and supports for the revised dynamic loads.

Piping stress reanalyses performed for the BOP piping due to power uprate changes were performed using the Bechtel ME101 analysis software, which is a later version of the same analytical computer code (ME632) used for the original analysis. See the response to NRC Question 8 for additional information regarding the Bechtel ME101 program.

Other BOP Components

PT calculations for power uprate were also used to evaluate other mechanical components such as manual valves, motor-operated valves, air-operated valves, solenoid-operated valves, bleeder trip valves, check valves, relief valves, tanks, heat exchangers, and pumps. Evaluations began by identifying changes in the PT calculations caused by the RSGs and power uprate. Line classes identified as having increased values were reviewed against the applicable piping and instrument drawings containing that line class. Mechanical components and valves within these line classes were then identified as requiring further evaluation due to increases in either design or maximum pressure and/or temperatures. The line classes identified were also used to evaluate impacts on insulation and room heat loads.

Components identified as requiring further evaluation were reviewed against the design conditions for the components. This included a comparison with such items as Code ratings, manufacturer's information, material used, etc. These were very detailed reviews which ensure that all components are acceptable for operation under power uprate conditions. Additionally, feedwater heaters were evaluated by the vendors for thermodynamic performance under power uprate conditions.

The results of these reviews were presented in Section 2 of the PULR, particularly in Section 2.4.2, "Main Steam Supply System," and Section 2.4.5, "Condensate and Feedwater."

Various computer codes were used to assist in these analyses. Bechtel FLASH TE605 was used for control valve sizing. PIPEFLOW was used for modeling the condensate and feedwater system, including the heater drain portion. The ASME B&PV Code, Section VIII, 1992 edition was used for relief valve sizing. The Heat Exchange Institute standard for closed feedwater heaters and a standard Napier formula for orifices were used to analyze feedwater vents to the condenser.

NRC Question 10

Provide the calculated maximum stresses for the critical BOP piping systems, the allowable limits, the Code of record and Code edition used for the power uprate conditions. If different from the Code of record, justify and reconcile the differences.

ANO Response

Critical BOP piping systems consist of the main steam (MS) and main feedwater (MFW) headers both inside and outside the containment building.

Stresses for main steam and feedwater piping inside containment are described in the response to Question 8.

The main steam and main feedwater header piping located outside of the containment building remains qualified for the power uprate parameters because the changes in the pressure and temperature due to power uprate are bounded by existing analyses of record. This conclusion was documented in the piping reconciliation calculations for the main steam and main feedwater PT calculations. A review of the pressure and temperature values in the PT calculations for these line classes confirmed that there were no changes in applicable values as a result of power uprate. Because the analyses for the main steam and main feedwater header piping outside containment were not revised, calculated stresses, allowable limits, and the analysis code and code edition remain unchanged.

NRC Question 11

In reference to Section 2.4.5.3, you stated that the feedwater heaters have been evaluated for the power uprate condition for extractions, design pressures, pressure drops, and drain, tube and nozzle velocities. You also stated that feedwater heater vibration characteristics and shell-side relief valve capacities have been evaluated. The main steam and feedwater flow rates increase about 10 percent for the power uprate as shown in Table 3-1. Discuss the potential for flow-induced vibration in the main steam and feedwater pipe and the BOP heaters and heat exchangers following the power uprate.

ANO Response

Based on studies discussed in Section 2.4.5 of Enclosure 5 to the Power Uprate Licensing Report, the original condensate, main feed water (MFW), extraction and drain system piping is generously sized, and will have new flow velocities that are well within acceptable and recommended ranges. Because of this, it was concluded that the MFW header piping is not expected to experience unacceptable flow-induced vibration as a result of changes from power uprate. The feedwater heaters were reviewed by the vendor and found to be acceptable. The feedwater trains are sized to carry a substantial load with a single train. This generous sizing renders the feedwater heaters less sensitive to flow-induced vibration. Any flow induced vibration problems caused by power uprate are expected to be confined to small vents and drains.

Historically, the main steam piping has been the system that has displayed the most sensitivity to flow-induced vibration. Because of this history ANO, along with a second party review by Southwest Research Associates, studied the potential changes in the main steam piping vibration due to changes in the pressure and mass flow rate for original design, (Cycle 14 - the last cycle with the OSGs), and the first cycle for power uprate (Cycle 16). This study evaluated the physical geometry of the piping, vibration data collected on the main steam piping, and the effects that pressure, temperature, and mass flow rate changes would have on the kinetic energy available to drive flow-induced vibration of the piping. A summary of the comparison between the original (Cycle 14) and post-power uprate conditions is provided in Table 11-1 below. From this table, it can be seen that although the mass flow rate is increasing, steam velocity and kinetic energy levels will be less after power uprate than during Cycle 14. Based on this and the measured vibration levels in the piping during Cycle 14, it is not anticipated that power uprate operating conditions will cause unacceptable increases in the vibration of the major main steam piping. Again, for the main steam system the only significant vibration problems are expected to be associated with small vent and drain piping. Modifications have already been installed to reduce the vibration levels on the level controllers on the 2E-1 (high pressure) feedwater heaters, and, as other unacceptable vibration levels are identified, appropriate measures will be taken.

Table 11-1

	Mass Flow Rate (lbm/hr)	**Pressure (psia)**	**Temp (°F)**	**Relative Flow Velocity**	**Relative Kinetic Energy**
Original Conditions	12697495	878.2	529.0	1.00 X Original	1.00 X Original
Cycle 14 Conditions	12720000	769.0	513.7	1.18 X Original	1.18 X Original
Power Uprate Conditions	13660920	900.0	532.0	1.05 X Original; 0.90 X Cycle 14	1.13 X Original; 0.96 X Cycle 14

However, it was recognized that there may be secondary effects that may not be predicted; therefore, ANO has implemented a piping walkdown and vibration testing program to identify and resolve any such problems. The pre-2R14 and post-2R14 walkdowns and testing identified the equipment and systems with potential vibration concerns. The start-up testing program included the installation of vibration monitoring instrumentation on the main steam piping inside containment, hand-held collection of vibration data on main steam piping outside containment, and structured walkdowns of virtually all of the piping outside containment for visual identification of piping vibration. For piping identified by visual walkdowns, vibration data collection was performed. As recommended by ASME/ANSI OM-3, "Operation and Maintenance Requirements for Preoperational and Initial Startup Vibration Testing of Nuclear Power Plant Piping Systems," any vibration exceeding the 0.5 inch per second screening criterion was evaluated by the Design Engineering Structural Group. This same testing approach is planned again for start-up after power uprate.

NRC Question 12

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. Confirm that safety-related motor-operated valves (MOVs) in your Generic Letter (GL) 89-10 MOV program at ANO-2 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 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.

ANO Response

For the safety-related pumps, minimum performance requirements did not increase due to power uprate. Therefore, power uprate has no effect on the functionality of the pumps. The safety analyses confirmed that these minimum requirements are sufficient for the pumps to perform their intended function. (Section 7 of the Power Uprate Licensing Report describes the safety analyses for power uprate.) Power uprate does not affect the ability of the pumps to meet their technical specification requirements. As stated in Section 2.4.6 of the Power Uprate Licensing Report, the emergency feedwater pumps are adequate for power uprate. Although decay heat will increase, engineering evaluations for power uprate determined that no change to the EFW pump flow rate is needed. Calculations demonstrate that the EFW pumps can provide the minimum flow rate necessary to support the safety analysis flow rate assumptions. Similarly, no changes are necessary for the high pressure injection pumps or the low pressure injection pumps.

As discussed in Section 5.9 of the PULR, the specific overpressure protection requirements of the ASME B&PV Code were evaluated for power uprate. All general requirements and component requirements for pressurizer safety valves and main steam safety valves were found to be in compliance with the code and the original design requirements. As discussed in Enclosure 4, Section 1.0.1 of our letter dated November 29, 1999 (2CAN119901), the pressurizer code safety valves' (PSVs) capacity ratings were revised based on the use of the Napier Factor which was adopted by later versions of the ASME B&PV Code. The revision supported rerating the PSVs. Analysis of bounding reactor and steam plant transients causing pressure excursions have been conducted. These transients were evaluated to ensure both peak primary and secondary pressure did not exceed 110% of design pressure. ANO-2 has no power-operated relief valves.

The safety-related motor-operated valves (MOVs) and air-operated valves (AOVs) were evaluated for the pressures and temperatures expected for power uprate. For systems not affected by power uprate (no pressure or temperature increase, no increase in differential pressure), the valves were considered acceptable without further evaluation.

The safety-related AOVs were evaluated and found to be acceptable for power uprate conditions. AOVs were evaluated for the proper pressure/temperature rating for the expected conditions. This included an evaluation of the air actuator for the same conditions as well as for the expected differential pressure which will be experienced by the valve under power uprate conditions. Of the safety-related AOVs for ANO-2, only the main steam isolation valves are in a system or application impacted by power uprate. These were determined to be capable of performing their intended function under power uprate conditions. Those AOVs performing a containment isolation function had been previously evaluated for the containment upgrade, which included power uprate conditions, and found acceptable.

The safety-related MOVs were determined to be acceptable for power uprate conditions. The MOVs, including the GL 89-10 MOVs, were evaluated for the pressures and temperatures expected for power uprate. The pressures and temperatures were reviewed with respect to each MOV's design function to stroke. As part of this review, applicable setpoint, maximum expected differential pressure (MEDP), seismic, and weak link calculations were examined for potential impact from power uprate. Based on this evaluation, no physical changes to MOVs are required for power uprate conditions.

In regard to Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," the required relief valves have already been installed. These relief valves have been evaluated for power uprate conditions and found to be acceptable.

Evaluations for Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," are scheduled to be completed by September 30, 2001. No significant impacts due to power uprate have been identified to date and none are expected.

NRC Question 13

Confirm whether the steam generator replacement and the proposed power uprate will increase the accident temperature, pressure and sub-compartment pressurization that affect the design basis analyses for steel and concrete in the containment, steam tunnel and the spent fuel pool. If the structural steel and concrete will be affected, provide the design basis margin and margins after considering increased accident loading due to the steam generator replacement/power uprate.

ANO Response

Before replacing the steam generators, ANO-2 evaluated the effect on containment of the replacement steam generators and a 7.5% power uprate. The revised loss of coolant accident (LOCA) and main steam line break (MSLB) analyses necessitated an increase to the containment design pressure to 59 psig. This was documented in correspondence dated November 3, 1999 (2CAN119903), "Proposed Technical Specification Change Request Supporting Containment Building Design Pressure Increase to 59 Psig." The effect of the replacement steam generators (RSGs) and power uprate on the accident temperature, pressure and compartment pressurization was included in that submittal. Enclosure 3 of the November 3, 1999, letter describes the LOCA and MSLB analyses which included the 7.5% power uprate. Enclosure 4 describes the structural reanalysis performed for 59 psig including the design basis analysis for the steel and concrete in the building. Enclosure 5 summarizes the review of structures, systems and components inside containment. This review included the compartment pressurization.

Section 9.2 of the Power Uprate Licensing Report discusses high-energy line break analyses evaluated for RSG and power uprate. Changes to these analyses have been incorporated into Amendment 16 of the ANO-2 Safety Analysis Report (SAR). SAR Section 3.6 discusses the ANO-2 main steam tunnel. Section 3.6.4.1.1.2 explains that only one break location inside the steam tunnel is postulated. This postulated break was reanalyzed for the environmental effects for a power level of 3026 MWt with credit for the flow limiting device located in each steam generator outlet nozzle. The peak pressure remains bounded by the previous evaluation. The peak temperatures were increased due to higher steam enthalpy conservatively predicted from superheating as steam passes over uncovered tubes. A new peak temperature of 424 °F near the end of blowdown at 190 seconds was calculated. However, the reinforced concrete wall of concern that separates the turbine building from the auxiliary building is a 3-hour fire rated barrier. A 3-hour fire rated barrier is designed to withstand temperatures well in excess of that postulated from a high-energy line break.

Cooling for the spent fuel pool was discussed in detail in our letter dated May 30, 2001 (2CAN050105). Pool temperatures will be maintained as they are currently. The cooling system is adequate for power uprate conditions. If spent fuel pool cooling is lost, the pool is allowed to boil and makeup is provided by the service water system; therefore, power uprate causes no increase in pool temperature under a loss of cooling condition. Since pool temperatures will not increase for normal operation or loss of cooling conditions, power uprate does not affect the design basis analysis for the steel and concrete in the pool.

Attachment 3

Licensee Identified Commitments for 2CAN080104

COMMITMENT	TYPE	
	One-Time Action	Continuing Compliance
Evaluations for Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," are scheduled to be completed by September 30, 2001.	✓	

Attachment 2 - Affidavit and Proprietary Responses

Affidavit and Proprietary Responses to the Mechanical and Civil Engineering Branch Request for Additional Information Regarding the ANO-2 Power Uprate

I, Norton L. Shapiro, depose and say that I am the Advisory Engineer of CE Engineering Technology, Westinghouse Electric Company LLC (WEC), duly authorized to make this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and described below.

I am submitting this affidavit in conjunction with the application by Entergy Operations Incorporated and in conformance with the provisions of 10 CFR 2.790 of the Commission's regulations for withholding this information. I have personal knowledge of the criteria and procedures utilized by WEC in designating information as a trade secret, privileged, or as confidential commercial or financial information.

The information for which proprietary treatment is sought, and which document has been appropriately designated as proprietary, is contained in the following:

- Enclosure 1 to letter LTR-OA-01-2, "Response to Questions 1, 2, 4, 6 and 7 of the Mechanical and Civil Engineering Branch Request for Additional Information Regarding the ANO-2 Power Uprate," August 21, 2000 (Specifically responses to Questions 1, 2, & 7)

Pursuant to the provisions of Section 2.790(b)(4) of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information included in the document listed above should be withheld from public disclosure.

i. The information sought to be withheld from public disclosure is owned and has been held in confidence by WEC. It consists of information concerning structural effects and structural analyses of the power uprate for Arkansas Nuclear One, Unit 2 (ANO-2).

ii. The information consists of test data or other similar data concerning a process, method or component, the application of which results in substantial competitive advantage to WEC.

iii. The information is of a type customarily held in confidence by WEC and not customarily disclosed to the public.

iv. The information is being transmitted to the Commission in confidence under the provisions of 10 CFR 2.790 with the understanding that it is to be received in confidence by the Commission.

v. The information, to the best of my knowledge and belief, is not available in public sources, and any disclosure to third parties has been made pursuant to regulatory provisions or proprietary agreements that provide for maintenance of the information in confidence.

vi. Public disclosure of the information is likely to cause substantial harm to the competitive position of WEC because:

   a. A similar product is manufactured and sold by major competitors of WEC.

   b. Development of this information by WEC required tens of thousands of dollars and hundreds of manhours of effort. A competitor would have to undergo similar expense in generating equivalent information. In order to acquire such information, a competitor would also require considerable time and inconvenience to perform required structural analyses and develop the associated analytical models.

c. The information consists of technical data and details concerning structural effects and structural analyses of the ANO-2 power uprate, the application of which provides WEC a competitive economic advantage. The availability of such information to competitors would enable them to design their product to better compete with WEC, take marketing or other actions to improve their product's position or impair the position of WEC's product, and avoid developing similar technical analysis in support of their processes, methods or apparatus.

d. In pricing WEC's products and services, significant research, development, engineering, analytical, manufacturing, licensing, quality assurance and other costs and expenses must be included. The ability of WEC's competitors to utilize such information without similar expenditure of resources may enable them to sell at prices reflecting significantly lower costs.

e. Use of the information by competitors in the international marketplace would increase their ability to market comparable analytical services by reducing the costs associated with their technology development. In addition, disclosure would have an adverse economic impact on WEC's potential for obtaining or maintaining foreign licenses.

Norton L. Shapiro

Norton L. Shapiro
Advisory Engineer

Sworn to before me this 21 day of August, 2001

Joan C. Hastings
Notary Public

My Commission expires: JOAN C. HASTINGS
NOTARY PUBLIC
MY COMMISSION EXPIRES SEP. 30, 2002