CP&L
A Progress Energy Company

James Scarola
Vice President
Harris Nuclear Plant

SERIAL: HNP-01-084
10CFR50.4

JUN - 4 2001

United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
REGARDING THE STEAM GENERATOR REPLACEMENT
AND POWER UPRATE LICENSE AMENDMENT APPLICATIONS

Dear Sir or Madam:

By letters dated October 4, 2000 and December 14, 2000, Carolina Power & Light Company (CP&L) submitted license amendment requests to revise the Harris Nuclear Plant (HNP) Facility Operating License and Technical Specifications to support steam generator replacement and to allow operation at an uprated reactor core power level of 2900 megawatts thermal (Mwt). NRC letter dated April 20, 2001 requested additional information to support staff review of the proposed license amendment requests. The requested information is provided in the Enclosures to this letter.

Please note Enclosure 2 to this letter contains information proprietary to Westinghouse Electric Company and is supported by an affidavit signed by Westinghouse, the owner of the information. Also enclosed is a Westinghouse authorization letter, CAW-01-1456 accompanying affidavit, Proprietary Information Notice, and Copyright Notice. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.790 of the Commission's regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations.

P.O. Box 165
New Hill, NC 27562

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A001

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse Affidavit should reference CAW-01-1456 and should be addressed to H. A. Sepp, Manager, Regulatory And Licensing Engineering, Westinghouse Electric Company, LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

The enclosed information is provided as a supplement to our October 4, 2000 and December 14, 2000 submittals and does not change the purpose or scope of the submittals, nor does it affect the conclusions of either the no significant hazards considerations or environmental evaluations previously submitted.

Please refer any questions regarding the enclosed information to Mr. Eric McCartney at (919) 362-2661.

Sincerely,

James Scarola

James Scarola
Vice President
Harris Nuclear Plant

James Scarola, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge, and belief, and the sources of his information are employees, contractors, and agents of Carolina Power & Light Company.

Darlene S. Yarborough

Notary (Seal)

DARLENE S. YARBOROUGH
NOTARY
***
PUBLIC
HARNETT COUNTY, NC

My commission Expires: 2-21-2005

KWS/kws

Enclosures (2)

c: Mr. J. B. Brady, NRC Senior Resident Inspector
Mr. Mel Fry, NCDENR
Mr. R. J. Laufer, NRC Project Manager
Mr. L. A. Reyes, NRC Regional Administrator

bc: (all w/Enclosure 1)

Ms. D. B. Alexander
Mr. G. E. Attarian
Mr. L. R. Beller (BNP)
Mr. C. L. Burton
Mr. J. R. Caves
Mr. H. K. Chernoff (RNP)
Mr. W. F. Conway
Mr. G. W. Davis
Mr. J. W. Donahue
Mr. R. J. Duncan II
Mr. R. J. Field
Mr. W. J. Flanagan
Mr. K. N. Harris
Ms. L. N. Hartz
Mr. C. S. Hinnant
Mr. J. W. Holt
Mr. M. T. Janus
Mr. W. D. Johnson
Ms. T. A. Hardy (PE&RAS File)
Mr. R. D. Martin
Mr. T. C. Morton
Mr. W. M. Peavyhouse
Mr. J. M. Taylor
Nuclear Records
Harris Licensing File (s) (2 copies)

NRC Question 1:

Please clarify if an assessment has been performed to confirm that the current plugging limit of 40 percent in the Technical Specifications (TS) will continue to be applicable for the replacement steam generators (SGs).

CP&L Response:

Westinghouse has performed a Regulatory Guide 1.121 analysis for the Shearon Harris replacement steam generators. The results of the analysis are summarized in WCAP-15678, "Regulatory Guide 1.121 Analysis for the Shearon Harris Replacement Steam Generators," dated May 2001. Based on the analysis results, the current plugging limit of 40 percent in the Technical Specifications will continue to be applicable for the replacement steam generators.

NRC Question 2:

Although not required by the TS, please clarify whether the licensee will be following the Electric Power Research Institute (EPRI) Pressurized-Water Reactor (PWR) Steam Generator Examination Guidelines for their pre-service and in-service inspection for the replacement SGs. If exceptions are taken to the EPRI Guidelines, please provide a summary of the scope for the pre-service inspection and the first in-service inspection.

CP&L Response:

HNP plans to follow the EPRI PWR Steam Generator Examination Guidelines for the pre-service examination of the replacement steam generators. To date, there have been no exceptions identified for the pre-service examination.

The existing Harris Nuclear Plant (HNP) philosophy has been to follow the in-service EPRI Steam Generator Examination Guidelines. HNP has followed the EPRI PWR Steam Generator Examination Guidelines for the recent past outage SG NDE examinations (no exceptions identified). The EPRI SG Examination Guidelines are currently in the process of being revised to incorporate recent industry issues. The revised EPRI document is scheduled for issuance to the industry by late 2001/early 2002. While the EPRI SG NDE Examination Guidelines are used during formulation of SG NDE in-service inspection work scope, it is premature to identify if there are exceptions to the EPRI Steam Generator Examination Guidelines until the revision is issued. The first in-service inspection of the replacement steam generators is approximately two years in the future.

NRC Question 3:

Please clarify the discussion of sleeving and F^* criteria starting on page 23 of Enclosure 1, "Basis for Proposed Changes," of the October 4, 2000, submittal. Since sleeving and F^* will be deleted, why are they included as a basis for the proposed changes?

CP&L Response:

Within Enclosure 1 of the October 4, 2000 submittal, the discussion of tube sleeving and F* repair criteria begins on page 1-23 following a subsection entitled Background. The Background subsection merely provides a brief discussion of current HNP Technical Specifications applicable to the Model D4 Steam Generators.

The subsection following the Background, which appears on page 1-24, is entitled Proposed Changes. The third paragraph of the Proposed Changes subsection states in part that the F* criteria, SG tube repair, and sleeve/preheater inspections are related to types of degradation associated with the Old Steam Generators (OSGs) and are not applicable to the Replacement Steam Generators (RSGs). This is reflected by the proposed change to Technical Specification 3/4.4.5.

Note: In letter SERIAL: HNP-98-089, dated August 24, 1998, the HNP response to an NRC request for additional information included a commitment to evaluate appropriate new and improved eddy current techniques, as they become available, for tubes repaired by sleeving. This commitment will not apply to the Model Delta 75 replacement steam generators (RSGs), however, since removal of the sleeving tube repair option is reflected in the Technical Specification change proposed by the October 4, 2000 submittal.

NRC Question 4:

Section 5.7.1.3 of Enclosure 6, "NSSS [Nuclear Steam Supply System] Licensing Report," of the October 4, 2000, submittal, states that the primary side and the secondary side components were evaluated for the effects of changes to the thermal transients due to the power uprate. Section 5.7.1.4.1 also states, on page 5.7-4, that "LOCA [Loss-of-Coolant Accident] effects, which would show some impact due to uprate, were evaluated for the effects of a loop pipe break. Since leak-before-break has been applied to HNP [Harris Nuclear Plant]. Only the large auxiliary line breaks need to be considered. Therefore, the current analysis is conservative and will continue to envelop the LOCA effects in the uprate condition."

1. Please clarify whether the thermal transient effects due to large-bore reactor coolant system (RCS) pipe break LOCAs were considered in the current licensing basis for the design of the HNP SGs. If not, explain why they were not considered (Note that the approved leak-before-break condition applies only to dynamic effects).

NRC Question 4 (cont.)

2. Please provide the stress analysis results for the primary side components of the SGs, including the SG tubes, to demonstrate the adequacy of the HNP Model Delta 75 replacement SGs for the effects of thermal transients arising from postulated large-bore RCS pipe break LOCA conditions during the power uprate.

CP&L Response:

1. Westinghouse has reviewed the RSG analyses and has determined that the application of leak-before-break has been appropriately used for the RSG analyses and assessments. As part of this review, Westinghouse has determined that the brittle fracture assessment required by Section III of the ASME Code, Subsection NB Paragraph NB-3211 (d) for the protection against brittle fracture for those ferritic parts of the RSG (Channel heads and tubesheets) that may be susceptible to brittle fracture should use LBLOCA inputs for temperatures and pressures. The use of the large auxiliary line breaks (small break LOCA) is not appropriate. The effect of temperature and pressure variations has been evaluated in the design of the HNP steam generators consistent with ASME Code requirements. All other analyses are local dynamic effects for which the use of LBB is appropriate. Particularly, for the S/G tube analysis, the dynamic effects of the Large LOCA were conservatively considered. The transient flow and pressure fluctuations following a postulated pipe break in one of the reactor coolant loops, as well as for the S/G shaking loads resulting from the large LOCA are considered in the S/G tube analysis.

2. Specific results are reported in the applicable component stress reports. Results that address the effects of LOCA report a maximum ratio of $K_I/K_{IC} < 0.6$ for a critical flaw size > T/4. Primary side components such as the divider plate or the tubes, composed of Inconel, will not fail in a brittle mode and therefore do not require a fracture mechanics evaluation to be performed.

NRC Question 5:

As a result of the SG replacement and power uprate, the feedwater system flow and pressure have to increase from those required for the current SGs at both the current and uprated power levels. Section 5.7.3 of the NSSS Licensing Report provides some discussion on the flow-induced vibration and wear. In general, the report states that the structural evaluation of the tubes addresses the effects of pressure and temperature resulting from the transients. However, no discussion is provided regarding the effects of increased flows on the flow-induced vibration of tubes.

Please discuss the potential for flow-induced vibration of the SG tubes due to various mechanisms, including, in particular, the fluid-elastic instability in the current SG at the current power level. Provide an evaluation of the flow-induced vibration of the SG tubes in the replacement SGs at the uprated power condition describing the analysis methodology, damping value of the tubes and the computer code used in the analysis. Provide the results of the predicted vibration levels during the normal operating condition and the worst-case transient condition, and the calculated fluid-elastic instability ratios. If the details of the analysis and the results are documented in a report, submit the report for staff review. Explain whether or not the analysis results are applicable to the degraded SG condition and why.

CP&L Response:

The design analysis of the HNP replacement steam generators had already included consideration of an Uprate. The change to the existing analysis was to account for an option for a lower maximum feedwater temperature of 375°F rather than the 440°F considered in the original design analysis. The result of the evaluation, as stated in the NSSS Licensing Report was that tube wear, for the case considering a maximum feedwater temperature of 375°F, was enveloped by the current analysis and is therefore acceptable.

Since the system parameters identified for the Uprated HNP were already considered as part of the design basis, there was no increased flow to be considered. The following is a summary of the analysis performed as part of the design analysis to address tube vibration/wear/corrosion at HNP.

Analysis Summary (Reference 1):

The possibility of tube degradation due to either mechanical or flow-induced excitation was considered. The evaluation included detailed analyses of the tube support system that references an extensive research program with tube vibration model tests. Consideration was given to potential sources of tube excitation including primary fluid flow within the U-tubes, mechanically-induced vibration, and secondary fluid flow on the outside of the tubes. The effects of primary fluid flow and mechanically induced vibration are considered to be negligible during normal operation. The primary source of potential tube degradation due to vibration is the hydrodynamic excitation by the secondary fluid on the outside of the tubes, and this area has been emphasized in both analyses and tests including evaluation of steam generator operating experience.

Three potential tube vibration mechanisms due to hydrodynamic excitation by the secondary fluid on the outside of the tubes have been identified and evaluated. These include potential flow-induced vibrations resulting from vortex shedding, turbulence, and fluidelastic 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 in 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 confirmed that vibration amplitudes would be acceptably small, even if the carefully controlled laboratory conditions were unexpectedly reproduced in the steam generator.

Flow-induced vibrations due to flow turbulence are also small: root-mean-square amplitudes are consistent with levels measured in operating steam generators with benign wear experience, and these vibrations cause peak stresses which are well below fatigue limits for the tubing material. Calculated stresses are provided for comparison with appropriate limits in the tube stress report. These stresses do not exceed [ ]^{a,c,e} ksi even when consideration is given to peak versus RMS distributions in combination with postulated gaps at supports adjacent to postulated clamped conditions (with appropriately reduced damping). Neither unacceptable tube wear nor fatigue degradation is anticipated due to secondary flow turbulence.

Fluidelastic tube vibration is potentially more severe than either vortex shedding or turbulence because it is a self-excited mechanism: relatively large tube amplitudes can feedback proportionally large tube driving forces if an instability threshold is exceeded. Tube support spacing incorporated into design of both the tube support plates and the anti-vibration bars in the U-bend region provides tube response frequencies such that the instability threshold is not exceeded for secondary fluid flow conditions for tubes which are effectively supported. This approach provides large margins against initiation of fluidelastic vibration for tubes that are effectively supported by the tube support system.

Small clearances between the tubes and supporting structure are required for steam generator fabrication. These clearances introduce the potential that any given tube support location may not be totally effective in restraining tube motion if there is a finite gap around the tube at that location. Fluidelastic tube response, constrained within available support clearances, is therefore theoretically possible if secondary flow conditions exceed the instability threshold when no support is assumed at the location with a gap around the tube.

This potential had been investigated both with tests and analyses for the U-bend region where secondary flow conditions have the potential to exceed the instability threshold if a tube does not contact provided supports as a result of fabrication tolerances. Tube vibration response is shown to have wear potential within available design margins for postulated bounding conditions which envelope fitup measured during fabrication as well as uncertainties in pertinent parameters. Wear analyses and supporting tests for limiting postulated fitup conditions include simultaneous contributions from flow turbulence and fliudelastic excitation. Best estimate wear calculations vary from [ ]^{a,c,e} inch for the small percentage of tubes not intimately in contact with supports at nominal clearances. The maximum stability ratio for all tubes for expected and design conditions is less than [ ]^{a,c,e}. Calculations considering tube/AVB fitup variability, flow uncertainty, uncertainties in wear calculation procedures (work rates and wear coefficients), and high steam flow transient effects using a 90 percent availability factor, yield a upper bound tube wear depth of [ ]^{a,c,e} inch versus a plugging margin that is 4 times higher.

Corresponding tube bending stresses are again well below fatigue limits as a consequence of vibration amplitudes constrained by available clearances. Maximum bending stresses do not exceed [ ]^{a,c,e} ksi even when postulated clamped conditions (again with reduced damping) adjacent to postulated gaps are considered. The corresponding fatigue usage is [ ]^{a,c,e} versus a limit of 1.0 when using the same lower bound fatigue curve (much more conservative than the ASME Code curve) previously used to correlate a rapidly propagating fatigue event for a conventional design.

Analyses and test therefore demonstrate that unacceptable tube degradation resulting from tube vibration is not expected for Shearon Harris Nuclear Power Plant replacement steam generators.

Additional Information – Vibration Analysis

<u>Design Conditions:</u>

Shearon Harris design conditions include about 30% margin in steam and feed flow over best estimate values for existing 100% power at steady state. This added margin used in establishing the design configuration, equivalent to about a 25% increase over best estimate values for 100% uprated power, is specified to provide margin for transient effects. When calculating bounds on potential tube fatigue and wear for postulated limiting support conditions, best estimate flows at uprated 100% power were used. Included is an explicit consideration of transient effects due to increased feed/steam flows in the evaluation to verify the design bases.

<u>Computer Code:</u>

Linear dynamic analysis was performed covering a range of support configurations for various tubes using the finite element code FLOVIB (Reference 2) written specifically for flow-induced vibration and fretting wear calculations for a multi-span structural member. Verification and qualification of FLOVIB for steam generator applications includes not only the analytical comparisons in configuration control files with the report, but also many comparisons with results form tests and operating steam generators. These include comparisons with a 49-tube test model of the inlet region in water flow, a quarter-scale model of the U-bend tested in air, a 0.01 power scale model steam generator, recent cantilever tube air flow tests, and operating Model 51, and Model F steam generators.

Damping:

Damping equal to [ ]^{a,c,e} percent of critical and conservative threshold instability constants of [ ]^{a,c,e} (straight-leg) and [ ]^{a,c,e} (U-bend) are reflected in subsequent calculations, except as modified to cover the postulated clamped boundary conditions. These cases reflect the conservative treatment of unexpected clamping of the tube at the 405 SS FDB/TSP/AVB structure as required. Damping values of [ ]^{a,c,e} and [ ]^{a,c,e} percent of critical are used for the FDB/TSP inlet and the top TSP, respectively, for these bounding conditions.

Applicability of Analysis to the Degraded Steam Generator Condition:

The replacement steam generators for HNP utilize thermally treated Alloy 690 tube material. Thermally treated Alloy 690 is considered to be the best available tubing material for nuclear steam generator applications. This material exhibits superior resistance to PWSCC degradation.

The Flow Induced Vibration analysis of the Delta-75 does consider various potential degraded tube conditions that could occur in the RSGs. The analysis considers corrosion/erosion allowances together with wear allowances and dimensional tolerances in the structural evaluation of the tubing. Nominal tubing dimensions and adjustment factors based on reduced stiffness are applied to account for corrosion/erosion/wear/tolerances during the ASME Code evaluation.

CP&L's continued assessment of any active degradation mechanisms that may occur in the HNP steam generators or similar steam generators throughout the industry will ensure adequate detection of the degradation mechanisms in the affected areas. Any new degradation mechanisms would also be detected by the planned inspections.

References

1. Westinghouse Report: WNEP-9719, Rev. 0, "Delta 75 Steam Generator Flow Induced Vibration and Tube Wear/Corrosion Evaluation", March 1998 (Westinghouse Proprietary Class 2C).

2. P. L. Busby, "Computer Program FLOVIB: Flow-Induced Vibrations of a Multi-span Structural Member", Westinghouse Electric Corporation Pensacola Report WNEP-8648, Rev. 1, 1990 (Westinghouse Proprietary Class 2).

Enclosure 2 to SERIAL: HNP-01-084

Documentation provided by Enclosure 2:

1. Proprietary Information Notice (1 page)

2. Copyright Notice (1 page)

3. Westinghouse letter "Application for Withholding Proprietary Information from Public Disclosure" (CAW-01-1456) with Affidavit CAW-01-1456. (7 pages)

4. Response to NRC RAI Question #5 (Westinghouse Proprietary Class 2). (4 pages)

Proprietary Information Notice

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.790 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) contained within parentheses located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.790(b)(1).

Copyright Notice

Enclosure 2 to SERIAL: HNP-01-084

Westinghouse letter "Application for Withholding Proprietary Information from Public Disclosure" (CAW-01-1456) with Affidavit CAW-01-1456.

Westinghouse Electric Company, LLC

Box 355
Pittsburgh Pennsylvania 15230-0355

May 22, 2001

CAW-01-1456

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

Attention: Mr. Samuel J. Collins

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: "CP&L Harris Nuclear Plant SGR/Uprating RAI" (Proprietary), May 2001

Dear Mr. Collins:

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-01-1456 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.790 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Carolina Power and Light.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-01-1456 and should be addressed to the undersigned.

Very truly yours,

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Enclosures

cc: S. Bloom/NRR/OWFN/DRPW/PDIV2 (Rockville, MD) 1L

P:DATA/DOCUMENTS/0562s

CAW-01-1456

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Henry A. Sepp, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC ("Westinghouse"), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Henry A. Sepp

Henry A. Sepp, Manager
Regulatory and Licensing Engineering

Sworn to and subscribed
before me this 22ND day
of May, 2001

Lorraine M. Piplica

Notary Public

LORRAINE M. PIPLICA
COMMONWEALTH OF PENNSYLVANIA
NOTARY PUBLIC

Notarial Seal
Lorraine M. Piplica, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Dec. 14, 2003
Member, Pennsylvania Association of Notaries

P:DATA/DOCUMENTS/0562s

(1) I am Manager, Regulatory and Licensing Engineering, in the Nuclear Services of the Westinghouse Electric Company LLC ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by the Westinghouse Electric Company LLC in designating information as a trade secret, privileged or as confidential commercial or financial information.

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

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system that include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in "CP&L Harris Nuclear Plant SGR/Uprating RAI" (Proprietary), May 2001 for the Shearon Harris Nuclear Plant, being transmitted by Carolina Power And Light letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk, Attention: Mr. Samuel J. Collins. The proprietary information as submitted for use by Carolina Power And Light for the Shearon Harris Nuclear Plant is expected to be applicable in other licensee submittals in response to certain NRC requirements for justification of other Westinghouse Model Δ75 replacement steam generator projects.

This information is part of that which will enable Westinghouse to:

(a) Provide Westinghouse Model Δ75 replacement steam generator design and stress information.

(b) Provide specific analyses or evaluation results related to the parameters that are being considered for the Shearon Harris Steam Generator Replacement/Uprating Project.

(c) Assist the customer to obtain NRC approval.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of meeting NRC requirements for licensing documentation.

(b) Westinghouse can sell support and defense of the replacement steam generator technology to its customers in the licensing process.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar licensing support documentation and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar design programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended for developing analytical methods and performing tests.

Further the deponent sayeth not.