Joseph E. Pacher
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Constellation Energy
Nuclear Generation Group

October 31, 2008

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: **R.E. Ginna Nuclear Power Plant**
Docket No. 50-244

Responses to Request for Additional Information Pertaining to Fourth 10-Year Inservice Inspection Program Relief Request Number 19

Reference: (1) Letter from J. Pacher, Ginna LLC, to NRC Document Control Desk, Subject: Fourth Ten-Year Interval Inservice Inspection Program Withdrawal of Relief Request Number 18 and Re-submittal of Relief Request Number 19, dated June 23, 2008.

(2) Letter from D. Pickett, NRC, to J. Carlin, Ginna LLC, Subject: Request For Additional Information Re: Fourth 10-Year Interval Inservice Inspection Program Relief Request No. 19 - R.E. Ginna Nuclear Power Plant (TAC NO. MD8733), dated October 3, 2008.

In Reference 1, R.E. Ginna Nuclear Power Plant, LLC (Ginna LLC) re-submitted a proposed code relief request associated with the deferment of reactor pressure vessel B-F weld examinations required by the Fourth Ten-Year Interval Inservice Inspection Program. On October 3, 2008 the NRC responded to that request with a request for additional information (Reference 2).

Enclosed are:
1. "Responses to NRC Request for Additional Information RE: Fourth 10-Year Interval Inservice Inspection Program Relief Request No. 19 - R.E. Ginna Nuclear Power Plant (TAC NO. MD8733)" (Proprietary)

2. "Responses to NRC Request for Additional Information RE: Fourth 10-Year Interval Inservice Inspection Program Relief Request No. 19 - R.E. Ginna Nuclear Power Plant (TAC NO. MD8733)" (Non-Proprietary).

Also enclosed is Westinghouse authorization letter CAW-08-2494 with accompanying affidavit, Proprietary Information Notice, and Copyright Notice. As Attachment 1 contains information proprietary to Westinghouse Electric Company LLC, it is supported 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 Commission and addresses with specificity the considerations listed in paragraph (b) (4) of Section 2.390 of the Commission's regulations.

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Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse affidavit should reference CAW-08-2494 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

No new commitments are being made in this letter.

Should you have questions regarding this matter, please contact David Wilson (585) 771-5219, or david.f.wilson@constellation.com.

Very truly yours,

Joseph E. Pacher

Attachment: (1) "Responses to NRC Request for Additional Information RE: Fourth 10-Year Interval Inservice Inspection Program Relief Request No. 19 - R.E. Ginna Nuclear Power Plant (TAC NO. MD8733)" (Proprietary)

(2) "Responses to NRC Request for Additional Information RE: Fourth 10-Year Interval Inservice Inspection Program Relief Request No. 19 - R.E. Ginna Nuclear Power Plant (TAC NO. MD8733)" (Non-Proprietary)

(3) Westinghouse Proprietary Affidavit

cc: S. J. Collins, NRC
D. V. Pickett, NRC
Ginna Resident Inspector, NRC

Attachment 2

Responses to NRC Request for Additional Information RE: Fourth 10-Year Interval Inservice Inspection Program Relief Request No. 19 – R.E. Ginna Nuclear Power Plant (TAC NO. MD8733) (Non-Proprietary)

Question 1

On page 2 of 5, Paragraph 5, the relief request states that "The subject examinations are currently scheduled to be performed during the Fall 2009 refueling outage." Please clarify why the inspections are being postponed. If relief is granted to postpone these 4th 10-year inservice inspection (ISI) interval inspections into the 5th 10-year ISI interval, what is your plan for completing the ASME Code required 5th 10-year ISI interval inspections of these reactor vessel nozzle welds? The staff notes that if the relief requested in ISI-19 is granted, the subject welds will need to be examined a second time during the 5th 10-year ISI interval unless specific relief is requested and granted.

Response

Relief Request Number 19 pertains only to the volumetric examination requirements of the six nozzle to safe end/elbow or pipe welds. The six volumetric examinations are requested to be postponed since they have normally been performed at the same time as the reactor pressure vessel weld examinations from the inside of the vessel. This method of examination from the inside of the vessel provides greater volumetric examination coverage as compared to performing these examinations from the outside surface. The current plans are to perform all of the reactor pressure vessel related examinations in the 2011 outage (related Relief Requests 18, 20, and 21).

If Relief Request Number 19 is granted to postpone the 4th 10-year volumetric examinations into the 5th 10-year ISI Interval, R.E. Ginna Nuclear Power Plant acknowledges that we are required either to perform the examinations again on applicable welds in accordance with the new 5th Interval ISI Program or to generate a new Relief Request. The new Relief Request would address the welds described within Relief Request Number 19.

Question 2

On page 2 of 5, Paragraph 5A, the relief request states "these welds are stainless steel welds that do not contain any Alloy 82 or 182 weld material." What are the material specifications for the welds, nozzles, and safe ends? Additionally, later in this same paragraph, the relief request states that "there have been no known incidents of cracking in non-Alloy 82/182 reactor vessel Category B-F welds." Please clarify whether there are no known incidents of cracking or there are no known incidents of structurally significant cracking of non-Alloy 82/182 reactor vessel Category B-F welds? The staff notes that a recordable/reportable indication was detected in a stainless steel weld in Class 1 piping of a pressurized-water reactor plant, albeit the weld was not a category B-F weld.

Response

The material specifications for the nozzles, welds, and safe-ends for the welds that are the subject of the relief request, are specified in Table 2-1.

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Table 2-1: Material Specifications					
	Nozzle	Weld Butter	Weld	Safe-End	Pipe

With respect to the statement regarding known incidents of cracking, there have been no known incidents of "structurally significant" cracking of non-Alloy 82/182 reactor vessel category B-F welds. As shown in Table 1 of the request for relief, no indications have been recorded in previous inservice inspections of these welds at R.E. Ginna. While indications have been recorded in the course of performing inservice inspections at other operating plants, these indications have not been determined to be structurally significant and are most often smaller than the acceptance standards of IWB-3500 of Section XI of the ASME Code. The indications that have been recorded were a result of fabrication, rather than having been service induced, since they were typically embedded flaws. This statement is supported by the fact that while issues such as primary water stress corrosion cracking of Alloy 82/182 have been the basis for various industry programs, there is no current industry program to address the cracking of these nozzles on a generic basis.

Question 3

Please provide a more detailed description of the 1999 inspections of each weld covered in the relief request including method used and coverage obtained. The staff understands that these nozzle welds have historically been inspected from the vessel interior due to accessibility issues; however, what coverage would be possible if the inspection were done from the outside diameter?

Response

The 1999 volumetric examination results are as follows:

Weld ID	Summary #	Description	
AC-1002-1	003600	Safe End to Nozzle (SI Line)	
Method	Results		Coverage
Volumetric(UT)	(FRAMATOME W-15) No Recordable		100%

Weld ID	Summary #	Description	
AC-1003-1	003300	Safe End to Nozzle (SI Line)	
Method	Results		Coverage
Volumetric(UT)	(FRAMATOME W-9) No Recordable		100%

Weld ID	Summary #	Description	
PL-FW-II	002100	Nozzle to Pipe (Buttered Weld)	
Method	Results		Coverage
Volumetric(UT)	(FRAMATOME W-7) No Recordable		100%

Weld ID	Summary #	Description	
PL-FW-IV	002700	Nozzle to Pipe (Buttered Weld)	
Method	Results		Coverage
Volumetric(UT)	(FRAMATOME W-13) No Recordable		100%

Weld ID	Summary #	Description	
PL-FW-V	002400	Elbow to Nozzle (Buttered Weld)	
Method	Results		Coverage
Volumetric(UT)	(FRAMATOME W-17) No Recordable		100%

Weld ID	Summary #	Description	
PL-FW-VII	003000	Elbow to Nozzle (Buttered Weld)	
Method	Results		Coverage
Volumetric(UT)	(FRAMATOME W-11) No Recordable		100%

The coverage that would be possible if the inspection was performed from the outside diameter is as follows:

Weld ID	Summary #	Description	
AC-1002-1	003600	Safe End to Nozzle (SI Line)	
Method	Limitation		Coverage
Volumetric(UT)	(weld obscured by concrete)		0%

Weld ID	Summary #	Description	
AC-1003-1	003300	Safe End to Nozzle (SI Line)	
Method	Limitation		Coverage
Volumetric(UT)	(weld obscured by concrete)		0%

Weld ID	Summary #	Description	
PL-FW-II	002100	Nozzle to Pipe (Buttered Weld)	
Method	Limitation		Coverage
Volumetric(UT)	(Sandbox Limitation)		50%

Weld ID	Summary #	Description	
PL-FW-IV	002700	Nozzle to Pipe (Buttered Weld)	
Method	Limitation		Coverage
Volumetric(UT)	(Sandbox Limitation)		50%

Weld ID	Summary #	Description	
PL-FW-V	002400	Elbow to Nozzle (Buttered Weld)	
Method	Limitation		Coverage
Volumetric(UT)	(Sandbox Limitation)		50%

Weld ID	Summary #	Description	
PL-FW-VII	003000	Elbow to Nozzle (Buttered Weld)	
Method	Limitation		Coverage
Volumetric(UT)	(Sandbox Limitation)		50%

Question 4

ASME Code, Section XI, Table IWB-2500-1, requires both volumetric and surface examinations for Category B-F, Item No. B5.10. What were the results of the surface examinations during the 3rd 10-year ISI interval? If the 4th 10-year ISI interval surface exams have not yet been performed, are they going to remain on schedule and be performed during the fall 2009 refueling outage? The staff notes the importance of surface examinations to provide some assurance that cracking is not present.

Response

The surface examination results during the 3rd 10-year (1999) ISI Interval is as follows:

Weld ID	Summary #	Description	
AC-1002-1	003600	Safe End to Nozzle (SI Line)	
Method	Limitation		Coverage
Surface(PT)	(weld obscured by concrete)		0% (1)

Weld ID	Summary #	Description	
AC-1003-1	003300	Safe End to Nozzle (SI Line)	
Method	Limitation		Coverage
Surface(PT)	(weld obscured by concrete)		0% (1)

Weld ID	Summary #	Description	
PL-FW-II	002100	Nozzle to Pipe (Buttered Weld)	
Method	Limitation	Results	Coverage
Surface(PT)	(Sandbox Limitation)	No Recordable	62% (1)

Weld ID	Summary #	Description	
PL-FW-IV	002700	Nozzle to Pipe (Buttered Weld)	
Method	Limitation	Results	Coverage
Surface(PT)	(Sandbox Limitation)	No Recordable	68.5% (1)

Weld ID	Summary #	Description	
PL-FW-V	002400	Elbow to Nozzle (Buttered Weld)	
Method	Limitation	Results	Coverage
Surface(PT)	(Sandbox Limitation)	No Recordable	75% (1)

Weld ID	Summary #	Description	
PL-FW-VII	003000	Elbow to Nozzle (Buttered Weld)	
Method	Limitation	Results	Coverage
Surface(PT)	(Sandbox Limitation)	No Recordable	72% (1)

Footnote (1): Third Interval Relief Request Number 36, Fourth Interval Relief Request Number 8

The Fourth Interval Inservice Inspection surface examinations associated with Relief Request Number 19 have not been performed at this time. Code Case N-663 has been incorporated within the R.E. Ginna Nuclear Power Plant Fourth Interval Inservice Inspection Program. The surface examinations associated with these welds are not required since they meet the criteria specified within Code Case N-663.

Question 5

On pages 4 and 5 of 5, Paragraph C states that flaw evaluations performed by Westinghouse for the Prairie Island and Point Beach Units can be used for R.E. Ginna because of comparable geometries and loading conditions. Provide a detailed comparison of piping geometries and loading conditions to demonstrate that the flaw evaluations for Point Beach and Prairie Island are applicable to R.E. Ginna.

Response

A comparison of dimensions for the R.E. Ginna, Point Beach, and Prairie Island nozzle geometries is provided in Table 5-1. As can be seen from Table 5-1, the R.E. Ginna nozzle dimensions match those for Point Beach exactly and vary by less than 10% from those for Prairie Island.

Table 5-1: Dimensional Comparison

Region/Dimension	R.E. Ginna	Point Beach	Prairie Island

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The stresses used in the development of the flaw evaluation handbook are a combination of transient thermal stresses, stresses as a result of normal operating loads exerted on the nozzle by the piping, and residual stresses. The residual stresses used in the flaw evaluation do not vary from plant to plant. The stresses that most impact the calculation of fatigue crack growth, and the acceptable life of a flaw, are the transient thermal stresses. These stresses vary over time, whereas the nozzle loads and residual stresses are constant. For this reason, a comparison of design basis transients and number of design basis occurrences is provided in Table 5-2. As can be seen from this table, the design basis transients for all three units are comparable to one another. The design basis transient that provides the greatest crack growth is heatup/cooldown, which has the same number of occurrences for all three plants. The design bases of Point Beach and Prairie Island include several transients that are not included in the R.E. Ginna design basis. Therefore, use of the Point Beach and Prairie Island design basis transients is bounding. It should also be noted that the pressure and temperature versus time profiles for the design basis transients are consistent for the three plants in this comparison. Since the geometric dimensions in Table 5-1 are similar, the maximum and minimum transient stresses will also be similar. Therefore, the transient stresses and number of occurrences used in the development of the Point Beach and Prairie Island flaw evaluation handbooks are bounding relative to their application to Ginna.

Table 5-2: Comparison of Design Basis Transients			
Transient	**Number of Design Occurrences**		
	R. E. Ginna	**Point Beach**	**Prairie Island**
Normal Conditions			
Heatup/Cooldown at 100°F/Hour	200	200	200
Load Follow Cycles (Unit loading and unloading at 5% of full power/min)	14,500	14,500	18,300
Step load increase and decrease of 10% of full power	2,000	2,000	2,000
Large step load decrease, with steam dump	200	200	200
Steady State fluctuations	Infinite	Infinite	Infinite
Upset Conditions			
Loss of load, without immediate turbine or reactor trip	80	80	80
Loss of power (blackout with natural circulation in the Reactor Coolant System)	N/A	40	40
Loss of flow (partial loss of flow, one pump only)	80	80	80
Reactor trip from full power	400	400	400
Faulted Conditions			
Large Loss of Coolant Accident (LOCA)	N/A	1	1
Large Steam Line Break (LSB)	N/A	1	1
Test Conditions			
Turbine roll test	N/A	10	10
Primary Side Hydrostatic test conditions	5	50	50
Cold Hydrostatic test	5	5	5

A comparison of nozzle loads is provided in Tables 5-3 and 5-4 for the reactor vessel inlet nozzle and outlet nozzle, respectively. These tables include maximum loads for the deadweight, normal thermal, and safe-shutdown earthquake (SSE) condition. The deadweight and thermal loads are used in the flaw growth evaluation, while the SSE loads are only used in the determination of the allowable flaw size. Loads for the safety injection nozzles were not readily attainable for the three plants. However, the inlet and outlet nozzle are more limiting in terms of flaw tolerance. The nozzle loads in the shear directions (F_Y and F_Z) are not included in these tables because they are not used in the development of the flaw evaluation handbooks. Likewise, the bending moments along the X-axis (M_X), which are the torsional loads, are not used in the flaw evaluation and are not included either. The M_Y and M_Z loads are combined to give the maximum bending moment M_B that is used to calculate the bending stress. As shown by Tables 5-3 and 5-4, while there are some differences from plant to plant, the nozzle loads for the three plants are comparable. Furthermore, as previously explained, flaw growth is most significantly impacted by the transient stresses. While the stresses from the nozzle deadweight and thermal loads of Tables 5-3 and 5-4 are added to the transient stresses, the difference between the minimum and maximum stresses, which drives flaw growth, is a function of the maximum and minimum transient stresses, which were discussed previously. The addition of the stresses from the nozzle loads in Tables 5-3 and 5-4 increases the "R" ratio used in the fatigue crack growth calculations, but the impact of this ratio is second order, compared to the impact of the difference in the maximum and minimum transient stresses. As can be seen from Tables 5-3 and 5-4, there are differences in the nozzle loads for Point Beach and Prairie Island, but these differences resulted in negligible differences between the flaw evaluation results in References 1 and 2, respectively. Therefore, it is expected that the differences in nozzle loads in Tables 5-3 and 5-4 for R.E. Ginna will have a negligible effect on the conclusions in the request for relief.

Table 5-3: Reactor Vessel Inlet Nozzle Loads

Load	R.E. Ginna	Point Beach	Prairie Island	Units

a,c

Table 5-4: Reactor Vessel Outlet Nozzle Loads				
Load	R.E. Ginna	Point Beach	Prairie Island	Units

a,c

Question 6

The licensee stated that the assumed flaw in the Point Beach and Prairie Island flaw evaluation will remain acceptable for at least 20 years. However, R.E. Ginna has more than 20 years of remaining service life between now and the end of the license renewal period. Therefore, discuss the exact period of time the assumed flaw will remain acceptable and why.

Response:

While Ginna has more than 20 years of remaining service life between now and the end of the license renewal period, the length of the requested extension of the inspection interval is less than 6 months. If the request to extend the interval were approved, then the time between inspections would be less than 11.5 years. The flaw evaluations referenced in the request for relief were performed using methodologies that have been approved in Section XI of the ASME Code to determine the operating life of a flaw found during an inservice inspection. These flaw evaluations show that even if a flaw of the size identified in the relief request (a/t=20% and a/l=6) was not detected during the last inservice inspection, it would not grow to the allowable flaw size in 40 years and would therefore remain acceptable until the end of the license renewal period. It is also important to note that this undetected flaw would still remain below the acceptable size for a time period well beyond the length of the requested extension in inservice inspection interval.

Question 7

The staff notes that walkdowns of the containment are completed during refueling outages to determine the presence of external leakage. Since this relief request is for the deferral of the volumetric inspections, what walkdowns will be done in the fall 2009 refueling outage to provide a high level of confidence that any leakage from these welds would be recognized? Also, what leakage detection systems are available to the operator for these weld locations.

Response

An ASME Section XI Class 1 system leakage examination will be performed with qualified VT-2 examiners at normal operating pressure and temperature following the 2009 refueling outage. The four primary nozzle weld areas will be observed through an opening in the bio-wall during the performance of the VT-2 examination. In addition, the areas where the four primary nozzles welds are located (e.g. sandboxes) were examined during the 2008 Refueling Outage. No leakage or evidence of leakage from these areas was noted. The initial boric acid walkdown during the 2009 refueling outage will also inspect the opening through the bio-wall for any boric acid build-up in the nozzle area.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. The particulate monitor (R-11) can detect a leak as small as 0.018 gpm within 20 minutes assuming the presence of noble gas decay products. The gaseous monitor (R-12) can detect a leak of 2.0 gpm to greater than 10.0 gpm within 1 hour and is considered a backup to the particulate monitor. The operators monitor the reactor coolant system (RCS) and containment for leakage through the use of procedures S-12.4, "RCS Leakage Surveillance Record Instructions", and S-12.2, "Operator Action in the Event of Indications of Significant Increase in Leakage".

Question 8

On page 5 of 5, Paragraph 6, Duration of Proposed Alternative, is not clear to the staff. (a) provide a specific end date for the relief request, (b) the staff notes that the end of the 4th 10-year ISI interval is scheduled for December 31, 2009. Therefore, please confirm that the ASME Code allowed extension would be effective until December 31, 2010, and (c) provide the start date of the 5th 10-year ISI interval.

Response

The end date of Relief Request Number 19 is May 30, 2011. The R.E. Ginna Nuclear Power Plant 4th Interval ISI Program will end December 31, 2009. The ASME Code allowed extension would be in effect until December 31, 2010 for ASME Category and Item Numbers that are identified within Relief Request Number 19. The 5th Interval ISI Program will start January 1, 2010 and end December 31, 2019.

References:

1. WCAP-11477, Revision 1, "Handbook on Flaw Evaluation for Point Beach Units 1 & 2 Reactor Vessels," July 1990.

2. WCAP-10363, "Handbook on Flaw Evaluation for Prairie Island Units 1 & 2 Reactor Vessels," December 1984.

Attachment 3

Westinghouse Proprietary Affidavit

Westinghouse

Westinghouse Electric Company
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Proj letter ref LTR-PCAM-08-46

Our ref: CAW-08-2494

October 24, 2008

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: "Responses to NRC Request for Additional Information RE: Fourth 10-Year Interval Inservice Inspection Program Relief Request No. 19 – R.E. Ginna Nuclear Power Plant (TAC NO. MD8733)," (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced relief request is further identified in Affidavit CAW-08-2494 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 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Constellation Nuclear.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-08-2494 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

cc: G. Bacuta (NRC OWFN 12E-1)

Enclosures

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared J. A. Gresham, 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:

J A Gresham

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed before me
this 24 day of October, 2008

Margaret L Gonano
Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Margaret L. Gonano, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Jan. 3, 2010
Member, Pennsylvania Association of Notaries

(1) I am Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, 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 rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 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 Westinghouse 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 2.390 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 constitutes 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 which 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 10 CFR Section 2.390, 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 "Responses to NRC Request for Additional Information RE: Fourth 10-Year Interval Inservice Inspection Program Relief Request No. 19 – R.E. Ginna Nuclear Power Plant (TAC NO. MD8733)" (Proprietary) for submittal to the Commission, being transmitted by Constellation Nuclear letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk.

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

(a) To support Constellation Nuclear's efforts to respond to the NRC request for additional information pertaining to the Fourth 10-Year Interval Inservice Inspection Program Relief Request No. 19.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar design information to its customers.

(b) The information requested to be withheld reveals the distinguishing aspects of a methodology or design which was developed by Westinghouse.

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 calculations 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 technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

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.

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