WOLF CREEK
NUCLEAR OPERATING CORPORATION

Terry J. Garrett
Vice President Engineering

January 18, 2008

ET 08-0005

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

Reference: 1) Letter ET 07-0004, dated March 14, 2007, from T. J. Garrett, WCNOC, to USNRC

2) Letter ET 07-0050, dated October 16, 2007, from T. J. Garrett, WCNOC, to USNRC

Subject: Docket No. 50-482: Response to Request for Additional Information Relating to Replacement of the Main Steam and Feedwater Isolation Valves and Controls

Gentlemen:

Reference 1 provided a license amendment request that proposed revisions to Technical Specification (TS) 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," TS 3.7.2, "Main Steam Isolation Valves (MSIVs)," and TS 3.7.3, "Main Feedwater Isolation Valves (MFIVs)." The Reference proposed changes to these specifications based on a planned modification to replace the MSIVs and associated actuators, MFIVs and associated actuators, and replacement of the Main Steam and Feedwater Isolation System (MSFIS) controls. Reference 2 provided responses to questions related to the Small Break Loss-of-Coolant Accident (LOCA) analysis that were provided by electronic mail on August 20, 2007.

The NRC provided by electronic mail on November 7, 2007, a second request for additional information based on a review of Reference 2. The responses to questions 1 and 2 are provided in Enclosure I as they contain Westinghouse proprietary information. WCNOC provided by electronic mail on December 18, 2007, a draft response to question 3 which proposed that question 3 be withdrawn. Based on discussions on January 9, 2008, with the NRC Project Manager, the NRC internal issues associated with question 3 have been resolved. However, it was requested that WCNOC provide the response to question 3 in this submittal.

A001
NRR

Enclosure I provides the proprietary Westinghouse Electric Company LLC LTR-LIS-07-898 P-Attachment, "Response to Second Round NRC Request for Additional Information on the Wolf Creek Small Break LOCA Analysis." Enclosure II provides the non-proprietary Westinghouse Electric Company LLC LTR-LIS-07-898 NP-Attachment, "Response to Second Round NRC Request for Additional Information on the Wolf Creek Small Break LOCA Analysis." As Enclosure I contains information proprietary to Westinghouse Electric Company LLC, it is supported by an affidavit signed by Westinghouse Electric Company LLC, 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 10 CFR 2.390 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 2.390 of the Commission's regulations. This affidavit, along with Westinghouse authorization letter, CAW-08-2369, "Application for Withholding Proprietary Information from Public Disclosure," is contained in Enclosure III. Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse affidavit should reference CAW-08-2369 and should be address to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company, LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

The additional information provided in the Attachments and Enclosures do not impact the conclusions of the No Significant Hazards Consideration provided in Reference 1. In accordance with 10 CFR 50.91, a copy of this submittal is being provided to the designated Kansas State official.

This letter contains no commitments. If you have any questions concerning this matter, please contact me at (620) 364-4084, or Mr. Richard D. Flannigan at (620) 364-4117.

Sincerely,

Terry J. Garrett

TJG/rlt

Attachment
Enclosures
- I - Westinghouse Electric Company LLC LTR-LIS-07-898 P-Attachment, "Response to Second Round NRC Request for Additional Information on the Wolf Creek Small Break LOCA Analysis"
- II - Westinghouse Electric Company LLC LTR-LIS-07-898 NP-Attachment, "Response to Second Round NRC Request for Additional Information on the Wolf Creek Small Break LOCA Analysis"
- III - Westinghouse Electric Company LLC CAW-08-2369, "Application for Withholding Proprietary Information from Public Disclosure"

cc: E. E. Collins (NRC), w/a, w/e
T. A. Conley (KDHE), w/a, w/e (Enclosure II only)
J. N. Donohew (NRC), w/a, w/e
V. G. Gaddy (NRC), w/a, w/e
B. K. Singal (NRC), w/a, w/e
Senior Resident Inspector (NRC), w/a, w/e

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Terry J. Garrett, of lawful age, being first duly sworn upon oath says that he is Vice President Engineering of Wolf Creek Nuclear Operating Corporation; that he has read the foregoing document and knows the contents thereof; that he has executed the same for and on behalf of said Corporation with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By ______________________
Terry J. Garrett
Vice President Engineering

SUBSCRIBED and sworn to before me this 18th day of January, 2008.

Gayle Shepheard
Notary Public

GAYLE SHEPHEARD
Notary Public - State of Kansas
My Appt. Expires 7/24/2011

Expiration Date 7/24/2011

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

Wolf Creek Nuclear Operating Corporation (WCNOC) letter ET 07-0004 provided a license amendment request that proposed revisions to Technical Specification (TS) 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," TS 3.7.2, "Main Steam Isolation Valves (MSIVs)," and TS 3.7.3, "Main Feedwater Isolation Valves (MFIVs)" based on a planned modification to replace the MSIVs and associated actuators, MFIVs and associated actuators, and replacement of the Main Steam and Feedwater Isolation System (MSFIS) controls. Letter ET 07-0050, dated October 16, 2007, provided responses to questions related to the Small Break Loss-of-Coolant Accident (LOCA) analysis that were provided by electronic mail on August 20, 2007. The NRC provided by electronic mail on November 7, 2007, a second request for additional information based on a review of letter ET 07-0050. Provided below is the response to the question 3 from the request for additional information (RAI). The responses to questions 1 and 2 are provided in Enclosure I.

3. *Regarding the response to Question 12., the comparison of the NOTRUMP code to the SEMISCALE Test S-UT-8 experiment is incomplete. The test was run out to 750 seconds; however, the NOTRUMP small break LOCA code simulation was only shown to 300 seconds. No liquid level nor two-phase level comparisons to the data were presented. No comparisons of the NOTRUMP clad temperature predictions with the test data were presented. The PCT occurs late (well after 300 seconds) in the event due to long term core uncovery. Because of the truncated comparisons, the impact of the NOTRUMP code COSI model changes on long term core uncovery, RCS pressure, and the attendant PCT behavior cannot be ascertained. Address how the COSI model can be considered to be properly validated if it is compared against a single integral test. Also provide (1) the NOTRUMP comparisons to S-UT-8 out to and including 750 seconds, (2) the liquid and two-phase level predictions in the core out to this time, and (3) the comparison of the NOTRUMP clad temperature prediction with the test data at the PCT location for this test.*

   It should be noted that other tests of particular importance which display long term core uncovery include SEMISCALE tests S-07-10 , S-07-10D, S-LH-1, and S-LH-2. The ROSA-IV small break LOCA integral tests provide additional data with long term core uncovery and heat-up. The SEMISCALE tests S-07-10 and 10D show long term core uncovery (including loop seal clearing effects) with PCTs near 1800 F during the late peak. There are also integral tests in the European data base that also display long term core uncovery and cladding heat-up.

Response: As discussed in letter ET 07-0004, the Small Break LOCA analysis was performed using the 1985 Westinghouse Small Break LOCA Evaluation Model including the use of the COSI condensation model in WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model." By NRC letter dated August 12, 1996, the Staff indicated that it had reviewed WCAP-10054-P-Addendum 2, Revision 1, and found "that the proposed safety injection (SI) steam condensation model is acceptable for referencing in NOTRUMP SBLOCA applications for operating reactors." Although the use of this methodology has been used by Westinghouse reactors for the past 11 years, the Staff has questioned the robustness of the existing test data comparisons utilized in the WCAP and previously approved by the NRC.

In October of 2005, the NRC staff attended a meeting at the Westinghouse offices which was a continuation of a series meetings that focused on the use of the NOTRUMP evaluation model for the Beaver Valley and Ginna Extended Power Uprate (EPU) Small Break LOCA analyses. The Staff had many questions including (but not limited to) mixture level swell, core noding, break spectrum, loop seal clearing, larger size Small Break LOCA's, etc. While in attendance of this meeting, the COSI model was questioned, which is the first time Westinghouse became aware of this. This questioning appeared to be borne from a statement presented in a technical paper. In this paper, the author observed the possible over-prediction of Emergency Core Cooling System (ECCS) condensation in certain CATHARE code simulations of the LOBI integral test facility. While Westinghouse considered this more a casual observance rather than a statement of fact, the Staff was not so inclined to agree. As such, informal requests were made of NOTRUMP simulations of the SEMI-SCALE facility in particular. Since Test SUT-08 had been run in the original licensing WCAP (10054) and is used in the NOTRUMP validation package, it was repeated with COSI both active and in-active. These results were provided informally to the Staff (February 2006) prior to the ACRS review of the Beaver Valley and Ginna applications for EPU. No further communication occurred in that time frame on this particular subject and both analyses were approved.

In May 2007, the Staff provided a draft RAI to D. C. Cook, Unit 1, related to their submittal of the Small Break LOCA reanalysis pursuant to 10 CFR 50.46. Question 4 pertained to use of the COSI steam condensation model and questioned the validation of the previously approved model.

> 4. The NOTRUMP version employed in the evaluation utilized the COSI steam condensation model. While the staff has previously approved this model, there have been no integral experiments validating this modification. Please provide the results of the COSI condensation model to integral small-break LOCA experiments with long-term core uncovery.

As a result of discussing the above question with D. C. Cook in a telephone conference on July 20, 2007 and subsequent internal deliberation, the Staff dropped draft Question 4 and issued the formal RAI (NRC letter dated August 10, 2007).

While question 4 to D. C. Cook is general in nature when compared to WCNOC question 3. WCNOC believes that question 3 is an invalid question based on the use an NRC approved evaluation model and no further information is being provided.

Westinghouse Electric Company LLC LTR-LIS-07-898 NP-Attachment, "Response to Second Round NRC Request for Additional Information on the Wolf Creek Small Break LOCA Analysis"
(NON-PROPRIETARY)

LTR-LIS-07-898 NP-Attachment

Response to Second Round NRC Request for Additional Information on the Wolf Creek Small Break LOCA Analysis

January 2008

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

RAI 1

The response to Question 8 does not appear to answer the question. Reference 1 showed that the limiting location for small break loss-of-coolant accident (LOCA) is the discharge leg of the reactor coolant pumps based on studies of breaks in the suction and hot leg piping. The RAI question is in regard to breaks in the discharge leg located at the top of the piping. The loop seal bottom elevation is 6.3 ft below the top elevation of the core. Therefore, during the long term following the largest small break that does not allow the reactor coolant system (RCS) to completely refill with ECC, the loop seals will eventually refill and remain full of liquid, which creates a large resistance to steam flow to the break. As such, the pressure differential between the core and broken cold leg will increase sufficiently to depress the two-phase level into the core for an extended period of time. While the peak cladding temperature (PCT) may remain below 2200 °F, oxidation limits can be exceeded. Because the Wolf Creek plant has deep loop seals, address the response of the Wolf Creek plant to breaks sizes where the RCS cannot be completely refilled with emergency core cooling (ECC) liquid. Breaks that cannot result in refill, and all larger breaks will have the potential for long term core uncovery.

RAI 1 Response

In order to demonstrate the effects of break orientation with the NOTRUMP Evaluation Model (EM), the limiting PCT transient for the Wolf Creek Generating Station (WCGS) Small Break LOCA (SBLOCA) analysis was re-run with the break oriented at the top of the RCS cold leg. [

]^{a,c} Since there is only minimal core uncovery for the break oriented at the top of the cold leg (Figure 1-2), clad heat-up calculations were not performed and as such no PCT or maximum local oxidation was calculated (i.e., results remain significantly below the acceptance criteria of 2200°F and 17%).

Small breaks may indeed undergo brief periods of uncovery as a result of loop seal re-plugging due to RCS system refill; however, this is highly dependent on the break orientation assumed, as well as ECCS performance characteristics utilized in the analysis. [

]^{a,c} As seen in the figures associated with the top break orientation, loop seal re-plugging was indeed observed (Figure 1-3) although the predicted uncovery associated with this type of phenomena was tenuous (Figure 1-2) and the core exit vapor temperatures observed were much less severe than predicted with a [ ]^{a,c} orientation (Figure 1-1).

Finally, only larger breaks which are oriented at the top of the RCS cold leg piping would be susceptible to the effects of loop seal re-plugging since a significant amount of water build-up in the cold leg piping is required prior to water (either from the broken loop or intact loop ECCS) being able to backflow into the loop seal piping. In a simplified sense, the RCS is acting as a manometer at this point. However, in order to establish this in a long term sense, a delicate set of conditions with regard to vent paths in the RCS which rely on lower steaming rates would need to occur. In the short term (i.e. prior to hot leg switch over) this would be extremely unlikely because of the high boil-off rates resulting from near term decay heat. As such, the loop seal plugging and purging would be an ongoing process that would not lead to any long term uncovery periods. This has been shown to be the case in facility tests such as ROSA.

In addition, it should be noted that the NOTRUMP-EM methodology does not consider the presence of the gaps between the core barrel upper plenum nozzles and the vessel. Although small, when considered around the entire circumference of the hot leg nozzles, the resulting flow area is not trivial. These gaps will be present since the hypothesized scenario dictates a condition where this area of the vessel will be filled with a saturated mixture and/or sub-cooled liquid such that temperature expansion concerns do not exist. It is quite likely that in the time frame assumed in the RAI, that this flow area, when coupled with the vessel upper bypass flow path, will provide enough vapor relief path to bypass the loop seal(s) entirely and eliminate the effects of loop seal plugging and re-plugging on core uncovery.

a,c

Figure 1-1: Wolf Creek 4-inch Break, Core Exit Vapor Temperature Comparison

a,c

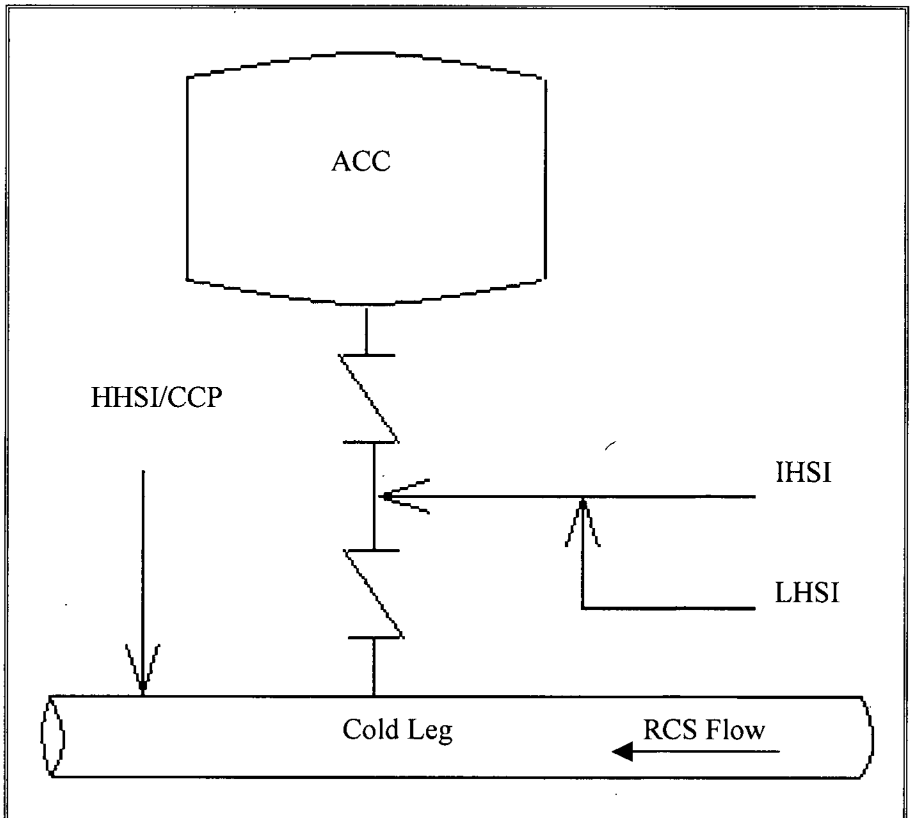
Figure 1-2: Wolf Creek 4-inch Break, Core Mixture Level Comparison

a,c

Figure 1-3: Wolf Creek 4-inch Break, Broken Loop Seal Vapor Flow Comparison

RAI 2
The response to Question 9 states that the high pressure safety injection (HPSI) line is attached directly to the cold leg. As such, address the following: (1) does a severed HPSI line with degraded HPSI flow cause PCT to become more limiting before the accumulators inject, (2) is there a break size on the RCS side of the break that delays accumulator and low pressure safety injection (LPSI) so only the degraded HPSI flow into the RCS occurs causing a more limiting PCT, and are the centrifugal charging pumps (CCPs) qualified for LOCA and actuated on the safety injection actuation signal (SIAS)?

RAI 2 Response
A diagram of the WCGS ECCS is provided below in Figure 2-1.



Legend	
ACC	Accumulator
HHSI/CCP	High Head Safety Injection/Centrifugal Charging Pump
IHSI	Intermediate Head Safety Injection
LHSI	Low Head Safety Injection

Figure 2-1: Wolf Creek Unit 1 ECCS Diagram

Part 1
As stated in the Round 1 RAI 9 response, the 8.75-inch equivalent diameter break modeled a severed accumulator line. Although realistically the HHSI would still inject to the broken loop cold leg, this scenario modeled all safety injection to the broken loop (Accumulator, HHSI/CCP, IHSI, and LHSI) to spill to containment pressure. For this case, there is only minimal core uncovery for a short duration. Furthermore, the core exit vapor temperature drops quickly to ~400°F and then steadily decreases to ~300°F over the remainder of the transient. Table 2-1 below summarizes the time sequence of events for

the 8.75-inch transient. Therefore, the severed accumulator line (and correspondingly, IHSI line) does not result in the limiting SBLOCA PCT for WCGS.

Table 2-1: Accumulator Line Break Time Sequence of Events

Event (sec)	
Transient Initiated	0.0
Reactor Trip Signal	7.16
Safety Injection Signal	8.77
Loop Seal Clearing Occurs	17
Safety Injection Begins(1)	47.77
Accumulator Injection Begins	175
RWST Low Level	1546.90

(1) Safety injection begins 39 seconds after the safety injection signal is reached.

Although the HHSI line is 1.5-inch schedule 160 piping, for breaks less than 8.75-inch (i.e., assumed to be a severed HHSI line), HHSI flow to the broken loop is assumed to inject to the RCS, not spill to the containment floor. Justification for this approach can be found in WCAP-10054-P-A, Addendum 2, Revision 1 (Reference 1). This is the first implementation of safety injection in the broken loop for WCGS. Results for the 2, 3, 4, 6, and 8.75-inch breaks were presented on page 64 of Attachment 1 in Reference 3; the PCT occurs after start of accumulator injection for the 4- and 6-inch cases, and accumulator injection is not observed for the 3-inch case.

Part 2

Reference 2, submitted to the NRC in July of 2006, discusses the aspects of a refined break spectrum. As stated in Reference 2, future applications using the NOTRUMP-EM will analyze a refined break spectrum if the PCT is approximately equal to or greater than 1700°F, or if the PCT results are close to or greater than the corresponding Large Break LOCA PCT results. The limiting 4-inch break resulted in a PCT of 936°F which is much less than the 1700°F PCT threshold. Furthermore, the LBLOCA PCT based on the Appendix K BASH-EM is 2088°F (including 10 CFR 50.46 assessments), more than double the SBLOCA PCT. It is clear that WCGS is LBLOCA-limited and as such, further refinement of the analyzed SBLOCA break spectrum is not necessary to determine a break size that delays accumulator and low pressure safety injection (LPSI).

In addition, the accumulator initial pressure analyzed in the WCGS SBLOCA analysis was 583 psia, which reflects the minimum Technical Specification value of 585 psig minus an uncertainty of 2.41% of 700 psi. While the minimum value was analyzed, realistically, the nitrogen cover pressure in each accumulator may vary anywhere from 585 psig to 665 psig according to WCGS Technical Specifications SR 3.5.1.3. Keeping this in mind, nominal cover pressure in any one accumulator would result in earlier injection for all breaks sizes that depressurize to this point as well as injection for smaller breaks (less than 4-inch) where accumulator injection may not have been observed to occur in the SBLOCA analysis.

The WCGS Emergency Core Cooling System (ECCS) consists of three separate subsystems: centrifugal charging (high head), safety injection (intermediate head), and residual heat removal (RHR) (low head). Each subsystem consists of two redundant, 100% capacity trains. The centrifugal charging subsystem of the Chemical and Volume Control System (CVCS) play an integral part in the emergency core cooling requirements for accidents such as LOCA, rod ejection accident, loss of secondary coolant accident, and steam generator tube rupture. The centrifugal charging pumps installed in the subsystem, in conjunction with other systems, provide borated ECCS flow to the RCS upon receipt of a safety injection signal (SIS). The borated injection flow provides core cooling and negative reactivity to ensure that the reactor core is protected; however, only the cooling aspect of the injected flow is modeled in the SBLOCA analysis. The safety injection function of the CVCS is automatically actuated by a low pressurizer pressure SIS as discussed on page 58 of Attachment 1 in Reference 3.

The operability of the centrifugal charging pumps, as part of the ECCS, is assured by the Technical Specification LCO 3.5.2, and its associated surveillance requirements. Therefore, the centrifugal charging pumps are qualified for mitigating a postulated design basis LOCA during power operations.

RAI 3

Regarding the response to Question 12., the comparison of the NOTRUMP code to the SEMISCALE Test S-UT-8 experiment is incomplete. The test was run out to 750 seconds; however, the NOTRUMP small break LOCA code simulation was only shown to 300 seconds. No liquid level nor two-phase level comparisons to the data were presented. No comparisons of the NOTRUMP clad temperature predictions with the test data were presented. The PCT occurs late (well after 300 seconds) in the event due to long term core uncovery. Because of the truncated comparisons, the impact of the NOTRUMP code COSI model changes on long term core uncovery, RCS pressure, and the attendant PCT behavior cannot be ascertained. Address how the COSI model can be considered to be properly validated if it is compared against a single integral test. Also provide (1) the NOTRUMP comparisons to S-UT-8 out to and including 750 seconds, (2) the liquid and two-phase level predictions in the core out to this time, and (3) the comparison of the NOTRUMP clad temperature prediction with the test data at the PCT location for this test.

It should be noted that other tests of particular importance which display long term core uncovery include SEMISCALE tests S-07-10, S-07-10D, S-LH-1, and S-LH-2. The ROSA-IV small break LOCA integral tests provide additional data with long term core uncovery and heat-up. The SEMISCALE tests S-07-10 and 10D show long term core uncovery (including loop seal clearing effects) with PCTs near 1800 F during the late peak. There are also integral tests in the European data base that also display long term core uncovery and cladding heat-up.

RAI 3 Response

To be provided by WCNOC.

References

1. WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997.

2. LTR-NRC-06-44, "Transmittal of LTR-NRC-06-44 NP-Attachment, 'Response to NRC Request for Additional Information on the Analyzed Break Spectrum for the Small Break Loss of Coolant Accident (SBLOCA) NOTRUMP Evaluation Model (NOTRUMP EM), Revision 1', (Non-Proprietary)," July 14, 2006.

3. ET 07-0004, "Docket No. 50-482: Revision to Technical Specification (TS) 3.3.2, 'Engineered Safety Feature Actuation System (ESFAS) Instrumentation,' TS 3.7.2, 'Main Steam Isolation Valves (MSIVs),' and TS 3.7.3, 'Main Feedwater Isolation Valves (MFIVs)'," March 14, 2007. (Available on ADAMS under accession number ML070800193)

Westinghouse Electric Company LLC CAW-08-2369, "Application for Withholding Proprietary Information from Public Disclosure"

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Direct tel: (412) 374-4643
Direct fax: (412) 374-4011
e-mail: greshaja@westinghouse.com

Our ref: CAW-08-2369

January 7, 2008

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-LIS-07-898 P-Attachment, "Response to Second Round NRC Request for Additional Information on the Wolf Creek Small Break LOCA Analysis," (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-08-2369 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 Wolf Creek Nuclear Operating Company.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-08-2369 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,

BFMaurer

B. F. Maurer, Acting Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: J. Thompson (NRC O-7E1A)

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared B. F. Maurer, 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:

B F Maurer

B. F. Maurer, Acting Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed before me
this 7th day of January, 2008

Sharon L. Markle

Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Sharon L. Markle, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Jan. 29, 2011
[illegible]

(1) I am Acting 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 LTR-LIS-07-898 P-Attachment, "Response to Second Round NRC Request for Additional Information on the Wolf Creek Small Break LOCA Analysis," (Proprietary), for submittal to the Commission, being transmitted by Wolf Creek Nuclear Operating Company letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with the request for NRC approval of the Wolf Creek Small Break LOCA Analysis.

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

(a) Assist the customer in obtaining NRC approval of the Wolf Creek Small Break LOCA analysis.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for other plant-specific applications.

(b) Its use by a competitor would improve his competitive position in the design and licensing of a similar product.

(c) The information requested to be withheld reveals the distinguishing aspects of a methodology 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.

In order to conform to the requirements of 10 CFR 2.390 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) 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 2.390(b)(1).

Copyright Notice