WOLF CREEK
NUCLEAR OPERATING CORPORATION

Terry J. Garrett
Vice President, Engineering

October 16, 2007

ET 07-0050

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

Reference: Letter ET 07-0004, dated March 14, 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:

The Reference 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.

The NRC provided by electronic mail on August 20, 2007, a request for additional information related to the non Loss-of-Coolant Accident (LOCA) analyses provided in Attachment I of the Reference. Attachment I provides Wolf Creek Nuclear Operating Corporation's (WCNOC) responses to questions 1-7. The NRC provided by electronic mail on August 23, 2007, additional questions (8 through 16) that are related to the Small Break LOCA analyses provided in Attachment I of the Reference. The responses to questions 8 through 16 (excluding questions 10 and 11) are provided in Enclosure I as they contain Westinghouse proprietary information. Attachment II provides WCNOC's responses to questions 10 and 11.

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Enclosure I provides the proprietary Westinghouse Electric Company LLC LTR-LIS-07-646 P-Attachment, "Response to 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-07-646 NP-Attachment, "Response to 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-07-2340, "Application for Withholding Proprietary Information from Public Disclosure," is contained in Enclosure III.

The additional information provided in the Attachments and Enclosures do not impact the conclusions of the No Significant Hazards Consideration provided in the Reference. 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. Kevin Moles at (620) 364-4126.

Sincerely,

Terry J. Garrett

TJG/rlt

Attachment I - Response to NRC Request for Additional Information (August 20, 2007)
II - Response to NRC Request for Additional Information (August 23, 2007)

Enclosures I - Westinghouse Electric Company LLC LTR-LIS-07-646 P-Attachment, "Response to NRC Request for Additional Information on the Wolf Creek Small Break LOCA Analysis"
II - Westinghouse Electric Company LLC LTR-LIS-07-646 NP-Attachment, "Response to NRC Request for Additional Information on the Wolf Creek Small Break LOCA Analysis"
III - Westinghouse Electric Company LLC CAW-07-2340, "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
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 16th day of October, 2007.

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 (August 20, 2007)

The NRC provided by electronic mail on August 20, 2007, a request for additional information related to the non Loss-of-Coolant Accident (LOCA) analyses provided in Attachment I of the Wolf Creek Nuclear Operating Corporation (WCNOC) letter ET 07-0004. 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. Provided below are responses to the questions in the request for additional information.

1. *Discuss if the increase in the MFIV closure time from 5 to 15 seconds will cause the steam generator (SG) overfilling condition to occur during an increased feedwater flow event. If the overfilling condition is predicted to occur, provide information to show that any SG safety or relief valves that are credited in the analysis of the event are operable.*

Response: The steam generator (SG) overfilling condition is predicted to occur during an increased feedwater flow event with a MFIV closure time of 15 seconds. This does not represent a change from the current analysis of record (AOR), which also predicted an occurrence of the SG overfilling condition with an assumed MFIV closure time of 5 seconds. Note that the occurrence of the SG overfilling condition is primarily due to a step increase to 200% of the nominal feedwater flow to two SGs assumed in the analysis.

Generally, the SG atmospheric relief valves (ARVs) are not assumed to function during the accident unless their function would result in more severe consequences. For the feedwater malfunction events, which result in excessive feedwater flow to one or more SGs, cases are analyzed with both the SG safety and ARVs modeled to open in response to a SG pressure increase during the transient. Since the feedwater malfunction events are analyzed primarily for departure from nucleate boiling (DNB), the SG ARVs are modeled as operational to minimize departure from nucleate boiling ratio (DNBR). As increased feedwater flow can result in an increase in heat removal by the secondary system, a decrease in moderator temperature occurs. In the presence of a negative moderator reactivity feedback condition, a decrease in the moderator temperature results in an increase in core power (nuclear flux). This yields a more limiting DNBR.

The SGs are isolated by closing the associated MSIVs, and this would be accomplished prior to SG overfill. Thus, after overfill of the SG secondary side occurs, water will accumulate in the steamline up to the MSIV and will eventually be released through the SG ARVs and/or safety valves. It is noted that water release through the SG ARVs and safety valves could potentially result in failure of the valves. The most likely mode of ARV or safety valve failure is for the valve to fail to reseat completely. This valve failure case would result in continuous water or two phase flow through the failed valve and tends to maximize the amount of water/steam mixture released to the atmosphere. However, it is expected that the result of the offsite dose evaluation for this scenario will be bounded by the offsite radiological consequences for the design basis steam generator tube rupture (SGTR) forced overfill scenario.

2. *In the LAR, the maximum valve closure time for the new MSIVs and MFIVs is being extended to 15 seconds. Discuss the model for the closure of the MFIVs and MSIVs in the accidents to clarify whether the model allows valves to ramp closed linearly in the 15 seconds valve closure time, the valve closure is delayed for 15 seconds following with an instantaneous closure, or another model is being used, and provide a reference to the NRC safety evaluation (SE) that approved the model in the analysis of record (AOR). Has this model been changed from the model used in the AOR, the current licensing basis? If the model used in the current licensing basis has changed, address the adequacy of the valve closure model used in the re-analysis, including any tests made of MSIV and MFIV closure.*

Response: For the non-LOCA transient AORs, the MSIV/MFIV closure models allow the valves to either ramp closed linearly over the valve closure time, or close instantaneously after a specific time delay, considering information such as signal transmission and valve closure, and the direction of conservatism with respect to a specific transient.

The NRC safety evaluation for Amendment No. 99, dated April 4, 1996, approved the models for the closure of the MSIVs/MFIVs in the AOR.

For the feedline break (FLB) and the SGTR re-analyses, the MSIV/MFIV closure models are unchanged, with differences due to the extended closure time period (i.e., 15 seconds).

The Main Steam Line Break (MSLB) AOR MSIV closure model of an instantaneous closure after a specific time delay is changed to ramp closed linearly over the conservative but bounding closure time of 15 seconds for the re-analysis. This model change was selected to offset the penalty associated with the longer valve stroke time, considering that the assumed 15 second valve closure time represents a bounding conservative value with respect to the equipment design data validation.

3. *List the computer codes and analysis methods used in the accident re-analyses provided in the LAR that are different from the AOR, and reference the associated NRC SEs that approved the use of the codes and methods. Address if there has there been any changes in the codes and methods used in the accident re-analyses, and show how the restrictions or conditions in the NRC SEs approving the use of these new codes and methods have been met.*

Response: WCNOC utilizes an NRC approved RETRAN-02 methodology and used the RETRAN-3D code in the RETRAN-02 mode for the non-LOCA transient analyses supporting the license amendment request submitted by letter ET 07-0004, dated March 14, 2007.

The NRC approved the use of RETRAN-02 in the Safety Evaluation dated September 30, 1993, for the "Transient Analysis Methodology for the Wolf Creek Generating Station," (WCNOC Topical Report NSAG-006). The upgrade of the RETRAN model to RETRAN-3D was performed utilizing the WCNOC calculation process. A design verification report was completed that performed a validity assessment of the NRC Safety Evaluation (dated January 25, 2001) which was incorporated into the EPRI Topical Report NP-7450(A), Revision 5. The performance of the design verification report ensured that the conditions and limitations in the NRC Safety Evaluation were complied with and that the correct RETRAN-02 models contained in the RETRAN-3D were used. Specifically, Condition 40 of the NRC Safety Evaluation Report states:

> 40. Organizations with NRC-approved RETRAN-02 methodologies can use the RETRAN-3D code in the RETRAN-02 mode without additional NRC approval, provided that none of the new RETRAN-3D models listed in the definition are used. Organizations with NRC-approved RETRAN-02 methodologies must obtain NRC approval prior to applying any of the new RETRAN-3D models listed above for UFSAR Chapter 15 licensing basis applications. Organizations without NRC-approved RETRAN-02 methodologies must obtain NRC approval for such methodologies or a specific application before applying the RETRAN-02 code or the RETRAN-3D code for UFSAR Chapter 15 licensing basis applications. Generic Letter 83-11 provides additional guidance in this area. Licensees who specifically reference RETRAN-02 in their technical specifications will have to request a Technical Specification change to use RETRAN-3D.

WCNOC has an NRC-approved RETRAN-02 methodology and is using the RETRAN-3D code in the RETRAN-02 mode. The NRC Safety Evaluation for Amendment No. 165, dated August 29, 2006, discusses the use of the RETRAN 3-D code in the RETRAN-02 mode. None of the RETRAN-3D models listed in the limitations are being used.

4. *Identify the values of any parameters, other than the MSIV and MFIV closure times, used in the accident re-analysis that are different from those used in the AOR, and justify the values changed for those parameters.*

Response: The re-analyses incorporated updates to the RETRAN base deck, primarily supporting the use of the RETRAN-3D software in RETRAN-02 mode or the revised MSIV/MFIV closure times.

All the RETRAN non-LOCA re-analyses incorporate RETRAN base deck corrective actions, including material changes revising the pressurizer safety valve nominal setpoint from 2485 psig to 2460 psig, the reclose blowdown pressure from 2375.45 psia to 2351.7 psia, and the full open (3% accumulation) pressure from 2574.25 psia to 2548.5 psia, as supported by the NRC Safety Evaluation for Amendment No. 133, dated March 23, 2000.

Due to the age of the SGTR with Stuck-Open ARV AOR and consistent with the AORs for the other non-LOCA transients analyzed, the RETRAN base deck used in the re-analysis of the SGTR with Stuck Open ARV also incorporated changes due to a reduction in thermal design flow (see NRC Safety Evaluation for Amendment No. 99, dated April 4, 1996).

5. *The licensee, listed on page 17 of 133 of Attachment I to its application, 12 major assumptions used in the steam line break (SLB) re-analysis. Address what are the changes to these assumptions in the SLB re-analysis because of the LAR, and provide information for each changed assumption to show that it is adequate for use in the analysis of the limiting SLB in terms of reactivity addition, core DNB performance and system response.*

Response: The assumptions revised from the AOR are discussed below:

Assumption #1 – Consistent with the Westinghouse Safety Analysis Standard (SAS) 12.0, a Reactor Coolant System (RCS) flow value corresponding to the thermal design flow is assumed. The current reduced thermal design flow of 361,296 gpm was used instead of the design flow value of 374,000 gpm used in the AOR. SAS 12.0 indicates that an increase in the peak heat flux, associated with the assumption of a flow rate higher than the thermal design flow, would be offset by crediting the higher flow rate in the DNBR calculation.

Consistent with SAS 12.0, the initial RCS pressure is assumed to be equal to the nominal RCS pressure. This represents a change from the –30 psia pressure uncertainty assumed in the current AOR. SAS 12.0 indicates that for a large steamline break, the depressurization is driven by the break and is not significantly affected by the initial conditions. SAS 12.0 further indicates that sensitivity analyses have shown that core power, inlet temperatures and pressures are virtually identical to the FSAR case if either 2200 psia or 2300 psia is assumed for the initial pressure.

Assumption #8 -- Auxiliary Feedwater (AFW) is activated immediately as the transient begins. Based upon the latest plant specific flow rate calculations, the AFW is assumed to be 1360.13 gpm to the faulted SG and 699.62 to the intact SGs, for a total of 2059.75. This is different from the 2020 gpm total to the faulted SG assumed in the AOR.

Assumption #9 – Steamline isolation (MSIV) closure is assumed to occur beginning two seconds after a low compensated steamline pressure signal is received, with the MSIV closure occurring in a 15 second linear ramp. This differs from the AOR, which assumes the MSIV closes instantly at 10 seconds.

Assumption #10 – Feedwater is isolated when safety injection is initiated, with a two second time delay. Note that closure of the valves simply causes additional friction loss in the lines and the effect is to reduce flow. However, continuous addition of full feedwater flow tends to cause an increase in core power by decreasing the reactor coolant temperature. Therefore, the feedline flow is assumed to ramp instantaneously to zero at the end of the 15-second time delay.

6. *The licensee listed, on page 27 of 133 of Attachment I to its application, 7 major assumptions used in the re-analysis for a feedwater line break (FLB). Address what are the changes to these assumptions in the FLB re-analysis because of the LAR, and demonstrate that the assumptions are adequate for an analysis of the limiting FLB with respect to the peak pressure and pressurizer overfilling aspects. Also, justify the operability of any pressurizer safety or relief valves credited in the FLB re-analysis, if the pressurizer is predicted to experience a water-overfilling condition.*

Response: For this re-analysis, the pressurizer was not predicted to experience a water-overfilling condition. The assumption revised from the AOR is discussed below:

Assumption #4 – The initial SG water level in all SGs is at the nominal value, plus 5.7% of the narrow range span. In this analysis the initial SG water level is assumed to be 55.7% of range, a 5.7% increase from nominal to conservatively delay the reactor trip. This represents a change from the AOR, which assumed an initial SG water level of 50%. Note: This was a portion of the disposition of Westinghouse Nuclear Safety Advisory Letter (NSAL)-03-9, "Steam Generator Water Level Uncertainties", 9/22/2003

The feedline break event can potentially be affected by the initial SG level uncertainty; specifically, if the SG water level is increased, it will delay the trip on the SG low-low level, prolonging the RCS heatup and the pressurization prior to trip. This will also prolong the moderator feedback induced power increase (the beginning of life moderator temperature coefficient is positive).

7. *In the steam generator tube rupture (SGTR) re-analysis, on pages 42 through 45 of Attachment I to the application, the licensee listed 5 and 7 assumptions (pages 43 and 44 of 133) used for the re-analysis of an overfill scenario and stuck-open atmospheric relief valve (ARV) scenarios, respectively. Identify the above assumptions that are different from those used in the AOR for this event, and justify the differences.*

Response: The SGTR with Forced Overfill event scenario includes the postulated failure of the faulted SG AFW flow control valve to maximize AFW flow to the ruptured SG to force SG overfill and subsequent relief from its safety valve. Note: Radionuclide consequence analyses, consistent with those for the AOR, demonstrate that the SGTR with Forced Overfill scenario is the limiting scenario in the SGTR analysis, as compared to the SGTR with Stuck-Open ARV scenario.

The assumption revised from the AOR for the SGTR with Forced Overfill scenario is discussed below:

Overfill Scenario Assumption Step #2 states "...AFW flow to the intact SGs maintains the narrow range level between 6% and 50% as indicated in the emergency operating procedure EMG E-3, "Steam Generator Tube Rupture."" This represents a change as the AOR narrow range level maintained was between 4% and 50%. This revision was identified and evaluated under the WCNOC corrective action program, which determined that the effect of the revision was insignificant. Note: In Assumption Step #4, consistent with the current AOR and Table 4.5-1 of Attachment I to ET 07-0004, the statement should reflect that the primary depressurization is initiated at eight minutes following termination of the RCS cooldown.

The SGTR with Stuck-Open ARV re-analysis scenario, characterized by discharge of activity to the atmosphere via the SG ARVs, includes as a substantive change from the assumptions of the AOR, revised critical operator actions.

These revised critical operator action times, intentionally selected to be more conservative than those of the AOR, were based upon a combination of simulator demonstrations and conservative extrapolations developed from experience with the forced overfill event in procedure EMG E-3.

Consistent with Tables 4.5-1 and 4.5-2 of Attachment I to ET 07-0004, and the results presented in the figures, the assumptions should reflect the following:

The discharge of contaminated secondary fluid is maximized by assuming an ARV stuck-open for 21.53 minutes, as compared with the AOR assumption of 20 minutes. [Assumption #2, page 44 of Attachment I to ET 07-0004]

RCS depressurization is initiated 5.6 minutes after cooldown is complete, as compared to the AOR assumption of three minutes. [Assumption #5, page 44 of Attachment I to ET 07-0004] Note: The Table 4.5-1 value stated should be 5.6 minutes.

Safety Injection is terminated after a 9.8-minute delay, as compared with the AOR assumption of a three-minute delay. [Assumption #6, page 44 of Attachment I to ET 07-0004]

A comparison of the critical operator action times for the re-analysis and the AOR is presented in the following table:

Action	Critical Operator Action Times (min.)	
	Analysis of Record	Re-Analysis
Tube Rupture Begins	0.0	0.0
Reactor Trip	2.37	2.38
Identify/Isolate Faulted SG	22.4	26.38
Initiate Cooldown	36.9	56.28
Terminate Cooldown	50.4	65.4
Initiate Depressurization	53.4	71.00
Terminate Depressurization	56.06	75.42
Terminate SI	59.06	85.23
Equalize Press/Backfill Faulted SG	61.01	115.3

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION (August 23, 2007)

The NRC provided by electronic mail on August 23, 2007, additional questions (8 through 16) that are related to the Small Break LOCA analyses provided in Attachment I of the Wolf Creek Nuclear Operating Corporation (WCNOC) letter ET 07-0004. The responses to questions 8 through 16 (excluding questions 10 and 11) are provided in Enclosure I as they contain Westinghouse proprietary information. 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. Provided below are responses to questions 10 and 11 in the request for additional information.

10. *Address how the SI water temperature is maintained at 100°F, as noted in Table 4.6-1, on page 61 of 133 in Attachment I to the application.*

Response: Prior to entering recirculation mode, the Safety Injection water is drawn from the Refueling Water Storage Tank (RWST); therefore, the maximum RWST water temperature is modeled in the Small Break LOCA analysis.

Technical Specification (TS) 3.5.4, "Refueling Water Storage Tank (RWST)," specifies a maximum RWST water temperature of 100°F. The RWST water temperature is verified, per Surveillance Requirement (SR) 3.5.4.1, every 24 hours to be within the limits (i.e., ≥ 37°F and ≤ 100°F). This frequency is sufficient to identify a temperature change that would approach either limit and has been shown to be acceptable through operating experience. The surveillance is not required to be performed when ambient air temperatures are within the operating limits of the RWST. With ambient air temperatures within the band, the RWST water temperature should not exceed the limits.

11. *Address how the water in the accumulators is maintained at 120°F inside containment, as noted in Table 4.6-1, and what is the maximum containment temperature that has been measured at Wolf Creek Generating Station.*

Response: It would be expected that accumulators located inside containment would have water temperatures following general trends of the containment air temperature. Therefore, the maximum containment air temperature of 120°F, allowed by TS 3.6.5, "Containment Air Temperature," was used in the Small Break LOCA analysis for the accumulator water temperature. The maximum containment temperature that has been measured at the Wolf Creek Generating Station (WCGS) is 119.2°F.

As accumulator water temperatures are expected to vary greatly during plant operation and are difficult to measure directly, any estimation of typical accumulator water temperature during normal full power operation can only be inferred from containment temperature conditions. Containment temperatures during full power operation will vary considerably with location inside of containment, with the warmest locations being inside the biological shield near the Reactor Coolant System piping as well as at the higher elevations. The coolest locations are typically in the lower elevations outside of the biological shield. The accumulators are located in the lower elevations of the containment, and outside of the biological shield. The accumulator water temperature would be expected to trend with containment air temperature in the lower elevations of the containment outside of the biological shield.

Westinghouse Electric Company LLC LTR-LIS-07-646 NP-Attachment, "Response to NRC Request for Additional Information on the Wolf Creek Small Break LOCA Analysis" (NON-PROPRIETARY)

LTR-LIS-07-646 NP-Attachment

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

October 2007

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

8. *Identify the bottom elevation of the suction legs and the top elevation of the core. If the suction leg elevation is below the top of the core, what is the limiting peak cladding temperature (PCT) for breaks at the top of the discharge leg?*

<u>Response:</u>

The bottom elevation of the suction leg piping is: 15.7549 ft
The top elevation of the core is: 22.0778 ft
The top elevation of the cold leg discharge pipe is: 28.4922 ft
Note: All elevations are with respect to the bottom of the reactor vessel.

For Small Break LOCA events, the effects of break location have been generically evaluated as part of the application of the NOTRUMP Evaluation Model (Reference 3), where it was concluded from the break location study that a break in the Reactor Coolant System (RCS) cold leg was limiting. Additionally, the effects of break orientation were considered during the evaluation of Safety Injection in the Broken Loop and application of the COSI Condensation Model (Reference 1). This work concluded that a break oriented at the bottom of the RCS cold leg piping was limiting with respect to Peak Cladding Temperature (PCT) [

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9. *Identify the PCT for the worst case with a severed safety injection (SI) line.*

<u>Response:</u>

The 8.75 in equivalent diameter break modeled a severed accumulator line. For Wolf Creek Generating Station (WCGS), the Low Head Safety Injection (SI) injects into the Intermediate Head SI which injects into the accumulator line. The High Head SI injects directly into the cold leg. Therefore, a severed accumulator line break would still result in High Head SI in the broken loop; however, all broken loop SI flows, including High Head SI, were assumed to spill to containment. 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. As such, fuel rod heat-up calculations are not warranted and no PCT was calculated.

10. *Address how the SI water temperature is maintained at 100°F, as noted in Table 4.6-1, on page 61 of 133 in Attachment I to the application.*

<u>Response:</u>

Response to be provided by WCNOC.

11. *Address how the water in the accumulators is maintained at 120°F inside containment, as noted in Table 4.6-1, and what is the maximum containment temperature that has been measured at Wolf Creek Generating Station.*

Response:

Response to be provided by WCNOC.

12. *It is noted, on page 57 of 133 of Attachment I, that several updates have been made to the NOTRUMP-EM code since the previous analysis of record (AOR) was completed, including the use of the COSI condensation model and SI in the broken loop. It was stated that these updates have been incorporated in the SB-LOCA re-analysis presented in the LAR. Provide information describing the COSI condensation model and validating the use of the model against relevant integral test data.*

Response:

The COSI condensation model / safety injection in the faulted loop methodology is described in detail in Reference 1 and approved in the Safety Evaluation (attached to Reference 1). Aspects of this model, bases and methodology were also discussed with the Staff throughout the licensing process of the Beaver Valley and Ginna extended power uprates, including the November 7-9, 2005 audit at the Westinghouse Energy Center. Integrated validation of COSI in the NOTRUMP Code includes simulations made using the SEMI-SCALE SUT – 08 tests. The results of these were provided to the Staff in information exchanges in support of the 2005 audit.

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13. *Explain what is meant by note (2) in Table 4.6-5, page 64 of 133 in Attachment I, regarding loop seal clearing. The note states that "Loop seal clearing is considered to occur when the broken loop seal vapor flow rate is sustained above 1 lbm/s [one pound mass per second]." If the mass flow rate condition is met, describe how the code treats loop seal clearing and the attendant thermal hydraulics in the suction leg. Also, address the basis of the 1 lb/sec clearing criteria in the suctions legs in the note.*

Response:

The 1 lbm/sec vapor flow criteria generally equates to the time that enough liquid is purged from the Reactor Coolant Pump (RCP) suction cross-over leg to establish a vapor vent path from the core to the break. Because there may be instances where the loop seal momentarily re-plugs, the time listed may not necessarily indicate the final clearing time. The 1 lbm/s flow rate is rather arbitrary, in fact it could be much higher without making any difference since when the loop seal clears, the vapor flow through the RCP suction cross-over leg typically becomes significantly greater than 1.0 lbm/s in a rapid manner.

When the vertical section of the RCP suction elbow at the Steam Generator (SG) outlet drains down to the horizontal section of the loop seal, vapor flow through the suction cross-over leg is sustained. The mixture region is quickly swept out of the loop seal area, particularly in the vertical section of the elbow to the RCP inlet and the RCP casing itself. NOTRUMP treats this flow in a co-current manner, with the resultant mixture region in the horizontal section of the loop seal being stratified. The mixture mass is carried with the vapor into the faulted cold leg where it is discharged through the break. The amount of mixture mass carried away in this interval is approximately 6,000 lbm. The majority of this is from the RCP since the horizontal section of the RCP suction cross-over leg has a smaller volume (approximately 20 ft^3). [For break sizes less than 6 inches equivalent diameter, this would occur for the faulted loop only, since the loop seal restriction is applied to the RCP suction cross-over leg in the non-faulted loop representation as described in Reference 2, as approved. [

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14. *Address if the SBLOCA re-analyses credits the hot leg nozzle gaps and/or core barrel alignment key leakage paths.*

<u>Response:</u>

The hot leg nozzle gaps are not considered or modeled in the NOTRUMP Evaluation Model (References 1 through 3). In addition, the head/upper support plate/core barrel/vessel alignment pins are essentially sealed from any coolant flow paths and are also not modeled.

15. *Address if there are new model changes included in the SBLOCA re-analyses, besides the updates mentioned above to the NOTRUMP-EM code, and provide the explanation and justification for these changes.*

<u>Response:</u>

The analysis utilized NOTRUMP Code Version 39.0. This is the same code version used for all recent Small Break LOCA analyses (since July of 2003) including those supporting the Beaver Valley and Ginna extended power uprates. As discussed in a meeting with the NRC Staff on October 13, 2005, no changes have been made to the NOTRUMP EM codes other than those for error corrections, which have been reported through the 10 CFR 50.46 process. The only change not per the 10 CFR 50.46 reporting process was the implementation of the COSI condensation model (Reference 1) which was approved by the Staff in 1996 and is identified in the Wolf Creek Nuclear Operating Corporation License Amendment Request (Reference 4).

16. *Identify the number of loop seals that are cleared for the 3, 4, and 6 inch break sizes listed in Table 4.6-5, and provide the suction leg liquid level plots for these cases.*

<u>Response:</u>

Only one loop seal clears for the 3 and 4 inch breaks. This is due to the artificially imposed restriction in the NOTRUMP Evaluation Model to restrict steam flow through the lumped intact loop as to ensure that venting steam flow through the loop seal of the broken loop will occur first. Justifications for its conservative effect on calculations are described further in References 2 and 3.

Both the broken and lumped intact loop loop seals clear in the 6 inch break (note that loop seal clearing in the lumped intact loop represents that all 3 intact loop loop seals have cleared). For this break, the artificial restriction was removed since six inch and larger breaks typically have sufficient steam flow to vent through all loops for an extended period of time. The removal of this restriction is only acceptable if the broken loop loop seal clears prior to the lumped loop. In this case, the broken loop loop seal clears at 55 seconds into the transient and the lumped intact loop seal clears seconds afterwards.

The suction leg liquid level plots for both the broken loop and lumped intact loop are provided below in Figures 1 through 3 for the 3, 4, and 6 inch breaks respectively.

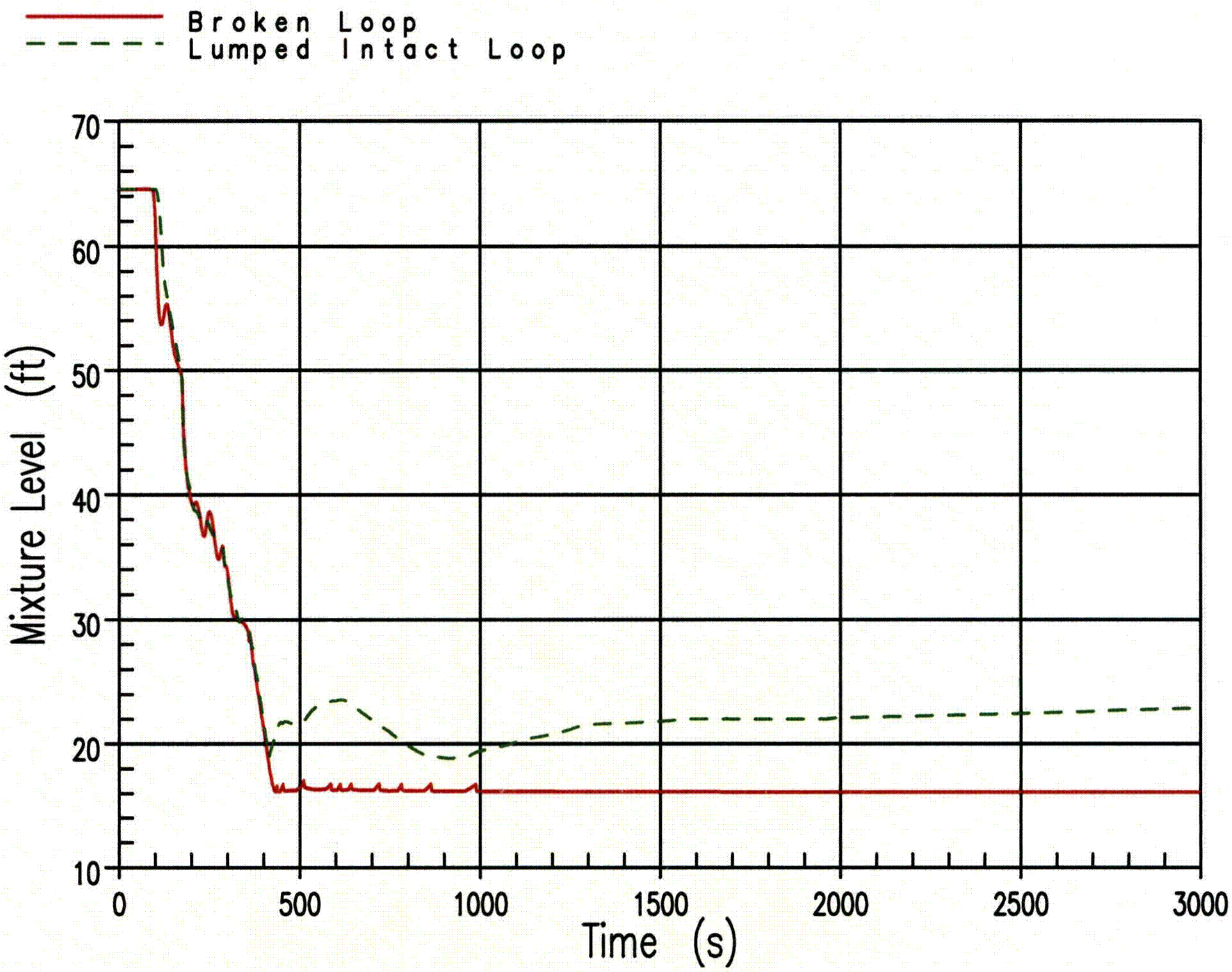

Figure 1: Suction Leg Liquid Level (3 inch break)

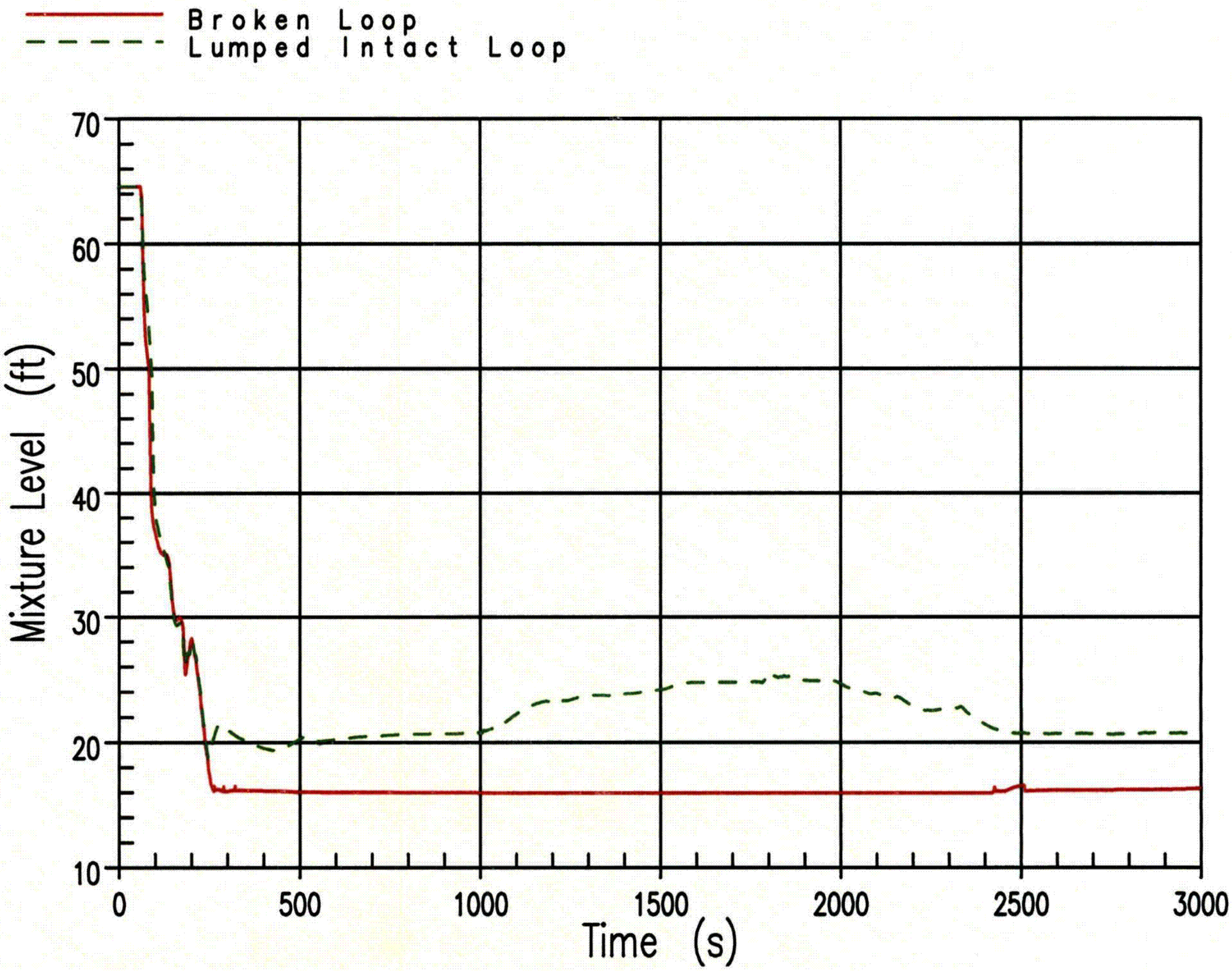

Figure 2: Suction Leg Liquid Level (4 inch break)

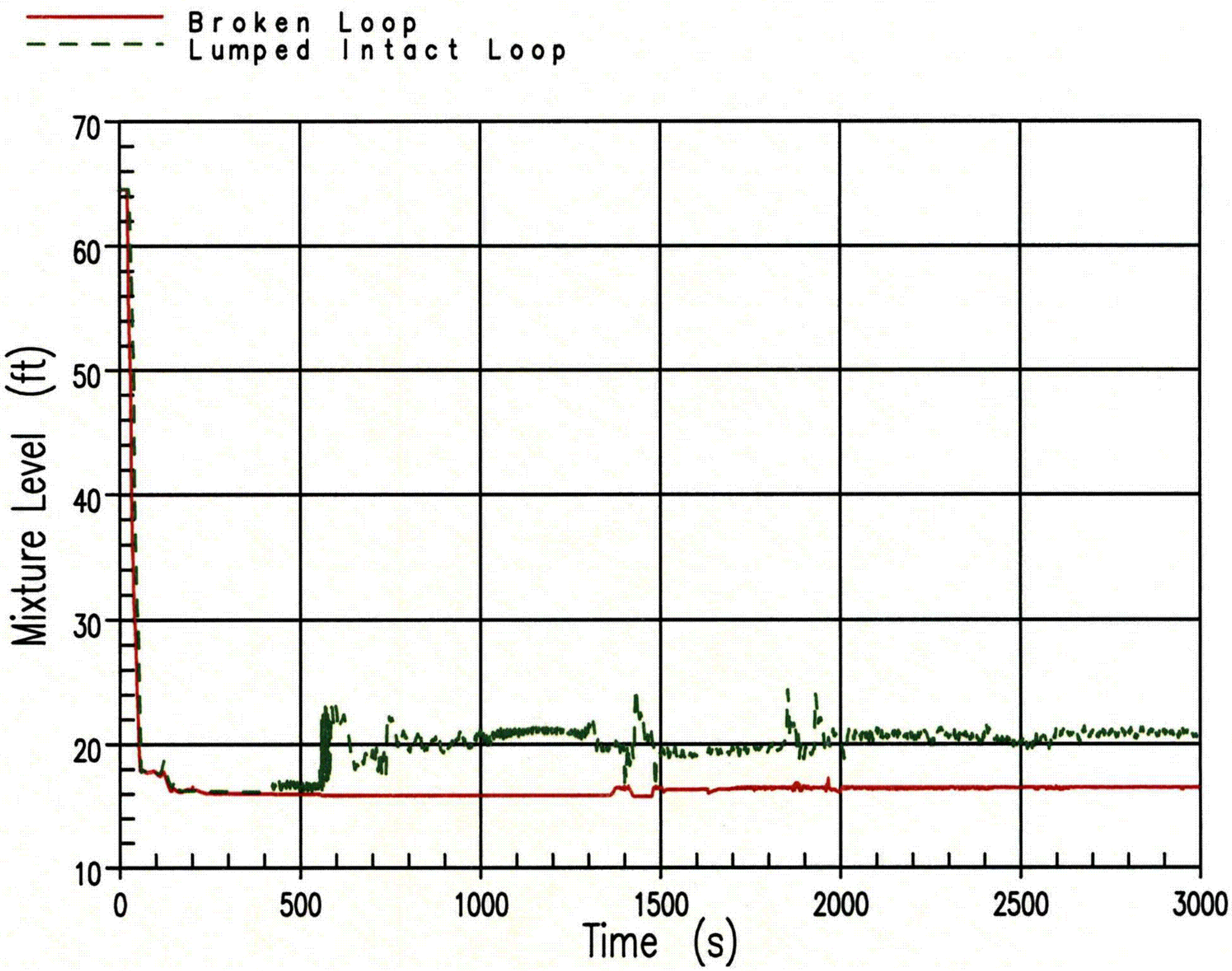

Figure 3: Suction Leg Liquid Level (6 inch break)

17. *Figure 4.6-2, page 65 of 133 in Attachment I, on the core mixture level is presented for the 4-inch break. Provide the core mixture level plots for the 3 and 6 inch breaks in the spectrum, where it is stated, on page 59 of 133 in Attachment I, that SBLOCTA code calculations were also performed.*

<u>Response:</u>

The core mixture level plots for the 3 and 6 inch breaks are shown below in Figures 4 and 5 respectively.

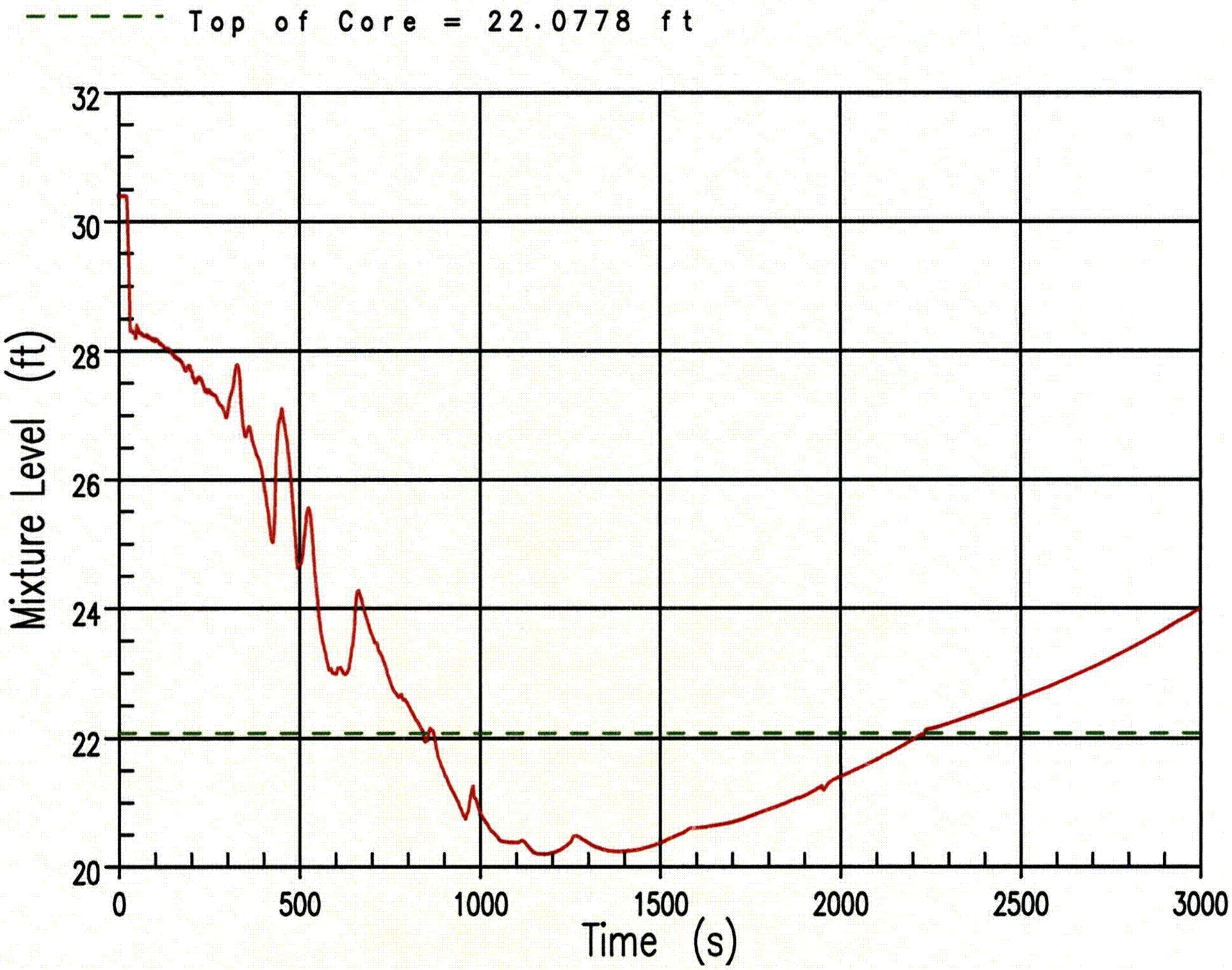

Figure 4: Core Mixture Level (3 inch break)

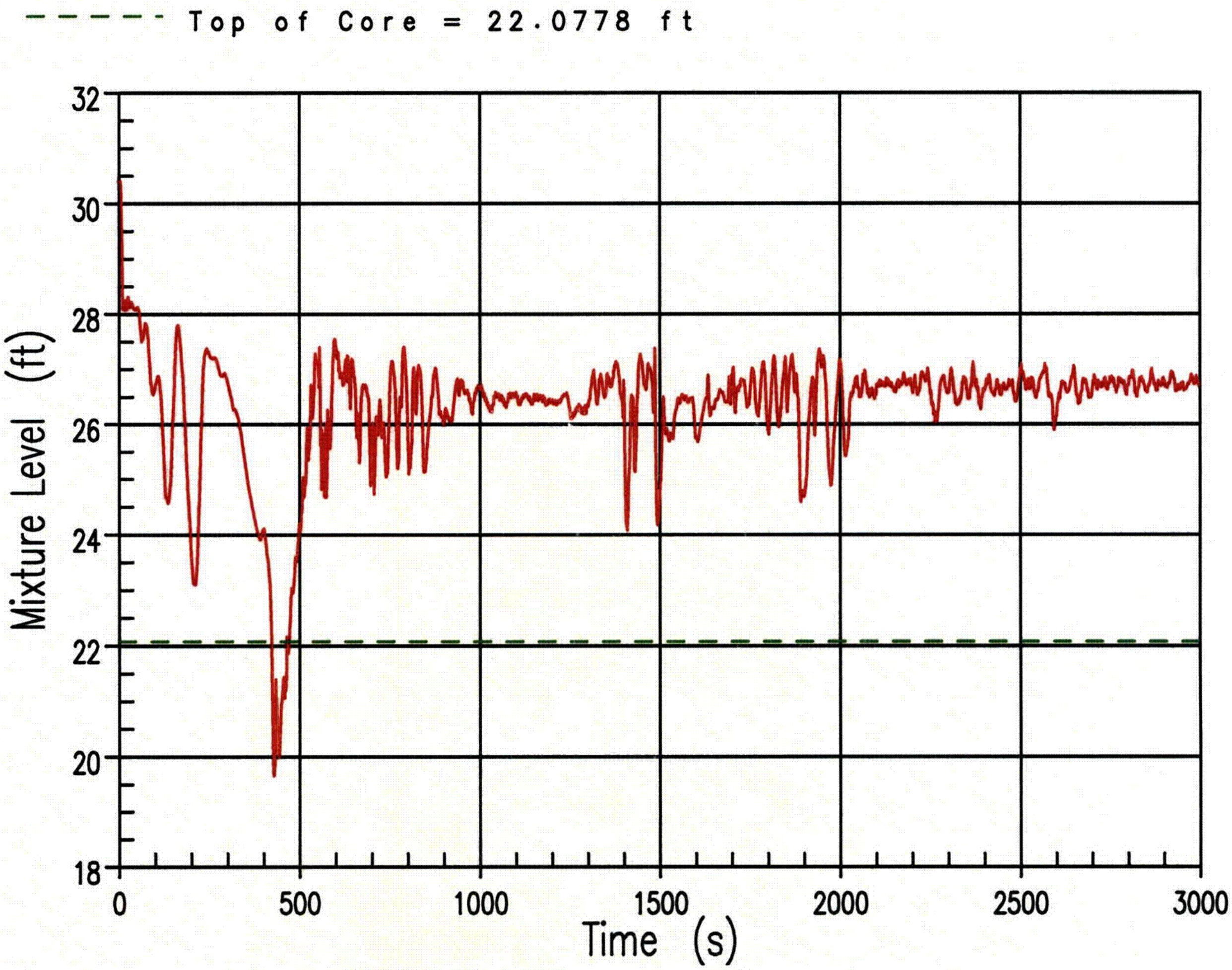

Figure 5: Core Mixture Level (6 inch break)

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. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.
3. WCAP-11145-P-A, "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code," October 1986.
4. 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.

Westinghouse Electric Company LLC CAW-07-2340, “Application for Withholding Proprietary Information from Public Disclosure”

Westinghouse

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Our ref: CAW-07-2340

October 8, 2007

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-LIS-07-646 P-Attachment, "Response to 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-07-2340 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-07-2340 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

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, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed before me
this 8th day of October, 2007

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
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 LTR-LIS-07-646 P-Attachment, "Response to 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).

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