Westinghouse Response to Supplemental NRC RAIs on WCAP-17524, "AP1000 Core Reference Report" (Non-Proprietary)

March 2013

Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, PA 16066

CRR-004-S1

In the average fuel assembly burnup sensitivity cases presented in response to RAI CRR-004, why did the PCT decrease for the hot rod when the overall stored energy increased?

Westinghouse Response to CRR-004-S1

In Figures 3A and 3B of the response to request for additional information (RAI) CRR-004 (Reference 1), it is observed that the sensitivity cases experienced slightly more cooling during the blowdown cooling period of the transient. This results in a slightly lower peak cladding temperature (PCT) at the end of blowdown. Also observed in the figures is that the heat-up rate during the refill and early reflood periods are equivalent between the base and sensitivity cases. With a slightly lower end of blowdown PCT and an equivalent heat-up rate to the time of PCT, the overall PCT is then a little lower.

The small additional blowdown cooling is a result of minor differences in the hot assembly liquid flow during blowdown due to minor differences in the core average and low power channel flows as a result of the increased core stored energy in those channels. The sensitivity cases have liquid entering the hot assembly slightly earlier during the blowdown downflow period, as demonstrated in Figure 1 for run069. The liquid reduces the vapor temperature during this period, which increases the heat transfer to vapor and ultimately reduces the cladding temperatures, as demonstrated in Figure 2 for run069. The figures also show that the differences in flow at the top of the hot assembly, and resulting earlier decrease in vapor temperature at the blowdown PCT elevation are short-lived, and that the general transient behavior is similar; this supports the conclusion that the global response is not substantially impacted and there is only a small impact on the calculated hot rod PCT.

a,c

Figure 1: Comparison of PCT and Total Mass Flow at the Top of the Hot Assembly Channel, run069

a,c

Figure 2: Comparison of Cladding Temperatures and Vapor Temperatures near the Blowdown PCT Elevation, run069

Reference

1. LTR-NRC-13-3, "Second Transmittal of Westinghouse Responses to NRC RAIs on WCAP-17524, 'AP1000 Core Reference Report' (Proprietary/Non-Proprietary)," January 10, 2013.

CRR-005-S1

The staff has the following questions regarding the response to RAI CRR-005:

a) What is the justification for allowing [ ]^{a,c}
If this is in the approved methodology, provide the reference.

b) Has PSHAPE ever been reviewed and approved by the NRC? If so, provide the reference.

c) Has the Westinghouse internal guidance allowing a [ ]^{a,c} of PBOT/PMID parameters ever been submitted for NRC review? If so, where is this information presented?

d) The section of the RAI CRR-005 response describing the PSHAPE routine presents [ ]^{a,c} When the PBOT, PMID, and peaking factors are transferred to PSHAPE, how is the resultant shape verified to be realistic?

Westinghouse Response to CRR-005-S1

(a) The check that the as-generated axial power shape has [ ]^{a,c}

(b) Use of the [ ]^{a,c}

(c) The Westinghouse internal guidance [ ]^{a,c}

[

]^{a,c}

(d) The axial power shape output from PSHAPE [

]^{a,c}

References

1. WCAP-12945-P-A, Volume 1, Revision 2, and Volumes 2 through 5, Revision 1, "Code Qualification Document for Best Estimate LOCA Analysis," March 1998.

2. WCAP-16009-P-A, Revision 0, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 2005.

CRR-006-S1

a) Previously, Westinghouse used a constant value of FQ based on the TS limit. If the accident analysis uses TS as the basis for the transient initiation, is Westinghouse planning on including the burndown curve as part of TS? If burndown credit for higher burnup regions of the core needs to be taken, this would mean the operation of the core would require the same restrictions.

b) In the response to RAI CRR-006, it is stated that [

]^{a,c}

   i. Provide justification to support that maintaining the hot assembly and hot rod at a single location is conservative.
   ii. Is the hot rod/assembly burnup a sampled parameter?
   iii. How is it ensured that the most limiting hot rod/assembly burnup has been sampled for a given case?

Westinghouse Response to CRR-006-S1

a) The plant-specific burnup dependent peaking factor limits (F_Q and $F_{\Delta H}$) used in the LBLOCA TCD evaluations were established with consideration of current operating cycles. The reduced peaking factors limits at high burnup conditions (or burndown limits) constitute a core design constraint and will be confirmed during the reload design process. The peaking factor burndown limits defined to support the LBLOCA peak cladding temperature (PCT) evaluations reflect the physical phenomenon of lower rod power as the fuel rod becomes less reactive, and are within the practical application of the core design. The plant-specific burndown limits were developed based upon review of a neutronic model's peaking factor behavior versus fuel rod burnup for actual cycle-designs for each plant. The burndown limit lines for the higher burnup fuel were defined such that there was slightly more margin (or "white space") to the predictive peaking factor data, as compared to that defined for the low burnup fuel (where no burndown credit is taken). By preserving more core design margin in the higher burnup region, the limit lines are more conservative for the purposes of a LBLOCA PCT evaluation. From a Technical Specification Surveillance perspective, this results in the lower burnup fuel to be more limiting, implicitly enveloping the higher burnup fuel. Therefore, no further Technical Specification Surveillance actions or COLR changes are required; the burndown credits supporting the LBLOCA PCT evaluations are confirmed analytically as part of the normal core design reload safety analyses process. An example of the burndown limits confirmation is shown in the following figures.

[

]^{a,c}

a,c

Figure 1: **AP1000** PWR F_Q Data[1] vs. Pin Burnup and Limits

Table 1: **AP1000** PWR F_Q vs. Pin Burnup Limits

a,c

[1] [ ]a,c
[2] [ ]a,c
[3] [ ]a,c

a,c

Figure 2: **AP1000** PWR $F_{\Delta H}$ Data vs. Pin Burnup and Limits

Table 2: **AP1000** PWR $F_{\Delta H}$ vs. Pin Burnup Limits

a,c

b)

i. In the **AP1000** ASTRUM analysis, the hot assembly is modeled as an assembly located [ ]^{a,c} This location was specified in accordance with the ASTRUM methodology (Reference 1). As discussed in Reference 1 Section 11-4-2, the limiting hot assembly location is chosen "*based on consideration of the hardware in the upper plenum, and the resulting flow distribution during the downward core flow period of blowdown. The selection of the limiting hot assembly location was supported by the responses to Requests for Additional Information (RAIs) 4-18 and 4-28 for 3- and 4-loop plants (appendix C, Part 3, of WCAP-12945-P-A*". The hot rod is modeled as within the hot assembly, in accordance with the ASTRUM methodology; see Reference 1 page 12-6.

The hot rod nominal power [ ]^{a,c} in the **AP1000** ASTRUM analysis which explicitly considers the effects of thermal conductivity degradation (TCD), peaking factor burndown is explicitly accounted for. Per the ASTRUM methodology, the hot assembly average LHR is assumed to be [ ]^{a,c} below the hot rod average LHR for each case; see Reference 1 page 12-9.

Therefore, the hot assembly/hot rod physical location is bounded by modeling the assembly in the location which minimizes effects of blowdown cooling. The hot rod and hot assembly power are conservatively high, in accordance with the ASTRUM method.

ii. [

]^{a,c}

[ ]^{a,c} (Equation 1)

[ ]^{a,c} (Equation 2)

Where:

[ ]^{a,c}

[

]^{a,c}

[ ]^{a,c}

[ ]^{a,c}

[

]^{a,c}

iii. The hot rod/hot assembly burnup treatment is described in part (ii).

Figure 3 shows the approximate range of maximum average fuel temperature for the hot rod initial condition in the large break LOCA analysis, based on the **AP1000** plant fuel rod design data with TCD and the FQ peaking factor burndown curves. From Figure 3, and considering other effects of TCD with burnup such as rod internal pressure changes, [

]^{a,c}

a,c

Figure 3: Approximate Range of **AP1000** Plant Hot Rod Maximum Average Fuel Temperature as a Function of Rod Burnup

Table 3: Representative Cycle Burnups and [ ]^{a,c}

a,c

(1) Beginning of cycle
(2) End of cycle

Reference

1. WCAP-16009-P-A, Revision 0, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 2005.

CRR-009-S1

"In response to CRR-009, Westinghouse stated that the AP1000 SBLOCA analysis provided in WCAP-17524-P does not result in any fuel rod heat up; therefore, no studies of the variations in rod internal pressures relative to burnup would be necessary. The staff notes that the SBLOCA analysis in WCAP-17524-P does not include thermal conductivity degradation (TCD) impacts."

"When considering TCD effects, how much fuel rod heatup is calculated for the AP1000 core and does this lead to significant changes to the previously calculated rod internal pressures during the SBLOCA analysis?"

Westinghouse Response to CRR-009-S1

During a Small Break Loss of Coolant Accident (SBLOCA), the fuel rods are surrounded by a two phase mixture of steam and liquid water for the majority of the transient. A core uncovery occurs if the level of this two phase mixture drops below the top of the active fuel rods during the transient. Stored energy in the core due to the initial fuel temperatures is not a significant effect in SBLOCA transients because the initial phase of the transient is gradual enough to remove the energy from the system, prior to any core uncovery. If predicted, core uncovery typically occurs later in the transient when there is insufficient makeup flow to replace the boil-off from decay heat and the result is fuel rod cladding heat up. Fuel rod internal pressures (RIPs) become important if there is significant fuel rod cladding heat up in a SBLOCA transient response; this is due to the impact of RIP on fuel rod cladding burst behavior.

The main effects of fuel thermal conductivity degradation (TCD) are increases in fuel temperatures and RIPs. An increase in fuel temperatures due to TCD leads to a higher amount of core stored energy. The **AP1000** plant SBLOCA analysis provided in WCAP-17524-P does not result in core uncovery and shows a significant amount of two phase mixture above the top of the active fuel; given that, and since the stored energy is removed during the early portion of a SBLOCA transient, the increase in core stored energy when considering the effects of TCD will not result in core uncovery (and subsequent fuel rod heat up) for any of the SBLOCA spectrum of breaks. An increase in RIPs due to TCD would have negligible impact on the **AP1000** plant SBLOCA analysis provided in Reference 1 since the core remains covered and there is no fuel rod heat up.

When considering the effects of TCD, increases in core stored energy and RIPs will not result in core uncovery for the **AP1000** plant SBLOCA analysis provided in Reference 1; therefore, fuel rod heat up will not occur.

<u>Reference</u>

1. WCAP-17524-P, "AP1000 Core Reference Report," March 2012.

CRR-026-S01

In response to RAI CRR-026, Westinghouse stated that a weighting method is used to give a more representative core average value. Describe the burnup and power weighting methodology including: (i) justification that this methodology is conservative to use for the initial, transitional, and equilibrium cores (ii) selection of the limiting burnup distribution, and (iii) selection of the limiting local power distribution.

Westinghouse Response to CRR-026-S1

Core stored energy (CSE) is defined as the amount of energy in the fuel rods in the core above the local coolant temperature. The Fuel Rod Design (FRD) group calculates CSE at a wide range of initial powers and rod average burnups. The limiting time in life for CSE when accounting for the effects of thermal conductivity degradation (TCD) is at [ ]^{a,c} as described in RAI CRR-026. CSE is a core average effect, and a large percentage of the core does not operate at these limiting conditions (i.e., [ ]^{a,c}). Therefore, conservative burnup and power weighting methods are used to determine the appropriate CSE.

The power and burnup methods described below were used initially in plant-specific TCD analyses for the current Westinghouse operating fleet. After performing evaluations for plants of different operating conditions and fuel types, it was determined that the weighting methods are applicable for a wide range of plant operation. A generic analysis was created to establish the applicability of this methodology for all plants, including the **AP1000** PWR. The generic power and burnup weighting methods are discussed below.

<u>Burnup Weighting</u>

First, the core stored energy values are sorted by burnup at each initial linear power level, as seen in the example table below. *Note that these values are used to better illustrate the burnup and power weighting methods and are not applicable to the **AP1000** PWR.*

a,c

[ ]

[

]a,c *Note that these values are used to better illustrate the burnup and power weighting methods and are not applicable to the* ***AP1000*** *PWR.*

a,c

[

]

[

]a,c

Power Weighting

[

]a,c *Note that these values are used to better illustrate the burnup and power weighting methods and are not applicable to the* ***AP1000*** *PWR.*

a,c

[ ]

Core Averaging

[

] a,c *Note that these values are used to better illustrate the burnup and power weighting methods and are not applicable to the **AP1000** PWR.*

a,c

[ ]

[

] a,c

a,c

[ ]

The power and burnup weighting methods described above provide a representative CSE value calculated using conservative assumptions ([ ] a,c) that remains bounding for a wide range of plant conditions throughout life.

CRR-027-S1

In response to RAI CRR-027, Westinghouse provided additional information regarding fuel assembly damping in flowing water.

a) In Figure 2 of the response, Westinghouse presents a proposed damping design curve in relation to the best fit of the presented test data. The staff requests that the best fit curve be adjusted to account for temperature effects and that the effects of measurement uncertainty and fuel design uncertainty be accounted for.

The response to RAI CRR-027 includes a discussion of the damping values as they decrease during a coast down.

b) The staff requests that the analysis be performed based on the tech spec minimum flow rate at the beginning of the transient.

Westinghouse Response to CRR-027-S1

The responses to these questions will be provided as supplemental information which will be provided to the NRC by April 30, 2013.

CRR-028

The core reference report for the AP1000 lists the CHF correlations which it will use to calculate the design DNBR limit using the RTDP methodology [WCAP-11397-P-A]. The Actual 95/95 DNBR limit is a function of the mean of the CHF measured to predicted data (μ), the standard deviation of the CHF measured to predicted data (б), and a statistical confidence factor which is only a function of the number of measured to predicted data points (k). However, the approved 95/95 DNBR limit is typically slightly higher than the actual 95/95 DNBR limit. For example, the mean (μ), standard deviation (б), and confidence factor (k) for the WRB-2M CHF correlation result in a 95/95 DNBR limit of [ ]^{a,c}. This limit is smaller than the approved 1.14 that has been rounded up to be more conservative.

The RTDP methodology statistically combines the uncertainties of the CHF correlation with other uncertainties in the DNBR calculation (e.g., flow, power, etc.). However, the RTDP methodology does not use the NRC Approved 95/95 DNBR limit, but the mean (μ), standard deviation (б), and confidence factor (k) directly. Using the values of the mean (μ), standard deviation (б), and confidence factor (k) will not result in the NRC approved 95/95 DNBR limit, but a value slightly lower than that limit (which is less conservative).

The staff is aware that the RTDP methodology does not require the calculation of the 95/95 DNBR limit, however in each case the staff's safety evaluations are written against that limit. Provide the 95/95 DNBR limit used in the RTDP methodology for each of the CHF correlations and demonstrate that the calculated limit is greater than or equal to the NRC approved 95/95 DNBR limit.

Westinghouse Response to CRR-028

Section 4.5 of the **AP1000** core reference report (Reference 1) lists the following applicable departure from nucleate boiling (DNB) correlations and their supporting references: WRB-2M (primary) (Reference 2), WRB-2 (Reference 3), ABB-NV (Reference 4), and WLOP (Reference 4). The 95/95 DNB ratio (DNBR) limits for the WRB-2M and WRB-2 correlations are a function of the mean of the critical heat flux (CHF) measured-to-predicted data ($m_{M/P}$), the standard deviation of the CHF measured-to-predicted data ($s_{M/P}$), and the Owen's one-sided confidence factor (K) (Reference 5) based on size of the database. The actual 95/95 DNBR limit was conservatively rounded up [ ]^{a,c} The correlation 95/95 DNBR limit is for DNB analyses where the uncertainty methodology used to satisfy the DNB design criterion is deterministic.

The conservative roundup [ ]^{a,c} is also applied to the 95/95 DNBR limit calculated using the Revised Thermal Design Procedure (RTDP) (Reference 6). RTDP convolutes the DNB correlation statistics with uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and computer codes to obtain the 95/95 limit. Applications of the correlation statistics to the RTDP limit calculation were illustrated in Section 3 of Reference 6 and in Tables 1 and 2 of Reference 7. Table 1 lists the WRB-2M and WRB-2 correlation statistics used in the **AP1000** PWR RTDP 95/95 DNBR limit calculations. As illustrated in Table 1, the calculated RTDP 95/95

WRB-2M limit has been rounded up [ ] ^{a,c} for the limiting hot channel. The similar roundup was also made to the RTDP 95/95 WRB-2 limit.

Since the ABB-NV DNB correlation limit was not based on the Owen's factor, its correlation statistics are not directly used with RTDP. [

] ^{a,c} To accomplish the adjustment for the **AP1000** PWR application, [
] ^{a,c}

The WLOP DNB correlation is not used with the RTDP for the **AP1000** PWR first core analysis. Because only a deterministic uncertainty methodology is used with WLOP, the 95/95 DNBR correlation limit is applied directly as approved in Reference 4.

In summary, similar conservative treatments have been applied to derivations of both the RTDP 95/95 DNBR limit and the correlation 95/95 DNBR limit. Since the WRB-2 and WRB-2M correlation limits were based on the Owen's one-sided confidence factor, the correlation statistics (M/P mean and standard deviation) were used in the **AP1000** PWR RTDP 95/95 DNBR limit calculation, consistent with the use of the correlation statistics in previous RTDP applications. The conservative roundup of the calculated RTDP DNBR limit [ ] ^{a,c} is consistent with the roundup for the NRC-approved correlation limit. [
] ^{a,c}

Table 1: DNB Correlation Statistics and RTDP DNBR Limits for **AP1000** PWR First Core Analysis

a,c

DNB Correlation	WRB-2M	WRB-2	ABB-NV
M/P Mean			
M/P Standard Deviation			
K (Owen's Factor)			
Calculated RTDP Limits			
95/95 RTDP Limit	1.25	1.27	1.19

[1] [
] ^{a,c}

[2] [ ] ^{a,c}

[3] [ ] ^{a,c}

References

1. WCAP-17524-P, "AP1000 Core Reference Report," March 2012.

2. WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids," April 1999.

3. WCAP-10444-P-A, "Reference Core Report - VANTAGE 5 Fuel Assembly," September 1985.

4. WCAP-14565-P-A, Addendum 2-P-A, "Addendum 2 to WCAP-14565-P-A, Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications," April 2008.

5. D. B. Owen, "Factors for One-Sided Tolerance Limits and for Variables Sampling Plans," SCR-607, March 1963.

6. WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989.

7. WCAP-12178-P-A, "Mini Revised Thermal Design Procedure (Mini RTDP)," October 1989.

Section H

Westinghouse

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LTR-NRC-13-26

April 30, 2013

Subject: Supplemental Information on End-of-Life Seismic/LOCA calculations for the AP1000 Pressurized Water Reactor (Proprietary/Non-Proprietary).

Enclosed are copies of the proprietary and non-proprietary versions of supplemental information related to the end-of-life Seismic/LOCA calculations for the **AP1000®** Pressurized Water Reactor (PWR). This supplemental information addresses the concern discussed in NRC information notice IN-2012-09 and the questions contained in the requests for additional information (RAIs).

Also enclosed is:

1. One (1) copy of the Application for Withholding Proprietary Information from Public Disclosure, AW-13-3704 (Non-Proprietary), with Proprietary Information Notice and Copyright Notice.
2. One (1) copy of Affidavit (Non-Proprietary).

This submittal contains proprietary information of Westinghouse Electric Company LLC. In conformance with the requirements of 10 CFR Section 2.390, as amended, of the Commission's regulations, we are enclosing with this submittal an Application for Withholding Proprietary Information from Public Disclosure and an affidavit. The affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference AW-13-3704 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

James A. Gresham, Manager
Regulatory Compliance

Enclosures

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AW-13-3704

April 30, 2013

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-NRC-13-26 P-Attachment, "Supplemental Information on End-of-Life Seismic/LOCA calculations for the AP1000 Pressurized Water Reactor" (Proprietary)

Reference: Letter from James A. Gresham to Document Control Desk, LTR-NRC-13-26, dated April 30, 2013

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC (Westinghouse), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.390, Affidavit AW-13-3704 accompanies this Application for Withholding Proprietary Information from Public Disclosure, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject 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 proprietary aspects of the application for withholding or the accompanying affidavit should reference AW-13-3704 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

James A. Gresham, Manager
Regulatory Compliance

Enclosures

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared James 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:

James A. Gresham, Manager
Regulatory Compliance

Sworn to and subscribed before me
this 30th day of April 2013

Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Anne M. Stegman, Notary Public
Unity Twp., Westmoreland County
My Commission Expires Aug. 7, 2016
MEMBER, PENNSYLVANIA ASSOCIATION OF NOTARIES

(1) I am Manager, Regulatory Compliance, 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 Proprietary Information from Public Disclosure 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-NRC-13-26 P-Attachment, "Supplemental Information on End-of-Life Seismic/LOCA calculations for the AP1000 Pressurized Water Reactor" (Proprietary), for submittal to the Commission, being transmitted by Westinghouse letter, LTR-NRC-13-26, 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 review of WCAP-17524, and may be used only for that purpose.

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

(a) Obtain NRC approval of the **AP1000**® Pressurized Water Reactor (PWR) Advanced First Core, as documented in WCAP-17524, "AP1000 Core Reference Report".

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of the information to its customers for the purpose of assisting customers in obtaining license changes for the **AP1000** PWR.

(b) This document establishes a portion of the licensing basis for the **AP1000** PWR.

(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 technical evaluation justifications 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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Supplemental Information on End-of-Life Seismic/LOCA calculations for the AP1000 Pressurized Water Reactor (Non-Proprietary)

April 2013

Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, PA 16066

1.0 INTRODUCTION

This report provides supplemental information to the **AP1000®** Core Reference Report (Reference 1) to address issues related to fuel assembly seismic structural response described in the NRC letter accepting this topical report for review (Reference 2). Specifically, the acceptance letter states:

> *"Fuel Assembly Seismic Structural Response – The Topical Report indicates (see Section 4.2.3.5) that the fuel assembly seismic response analysis follows the guidelines of Appendix A of the Standard Review Plan, Section 4.2, which may imply that only the beginning of life (BOL) condition of grid strength is needed for seismic and LOCA evaluations. Because of spacer grid spring relaxation due to irradiation which could affect the fuel bundle stiffness and the grid strength, the NRC staff has determined the need to evaluate the fuel structural response to the seismic/LOCA load for the minimum grid strength considering irradiation effects."*

To address this issue, Westinghouse has provided responses to three related RAIs (24, 25, & 27) in References 3 & 4 and performed end-of-life (EOL) tests and analyses. Basically, these tests and analyses are a repeat of the beginning-of-life (BOL) tests and analyses considering EOL effects. The RAI responses and the EOL tests and analyses are described in this report. The results demonstrate continued satisfaction of the acceptance criteria considering end-of-life (EOL) conditions.

Testing was performed to determine the grid strength and the fuel assembly dynamic characteristics at EOL conditions. The dynamic analyses to determine the EOL grid impact loads were performed using models and methods consistent with the BOL models and methods considering grid and fuel assembly properties and characteristics determined from the EOL tests. In addition, increased fuel assembly damping was used in the EOL core dynamic analysis compared to the damping value used in the BOL analysis. Justification for the fuel assembly damping value used in the EOL analysis is provided in this report and in the RAI responses.

2.0 GRID STRENGTH AT EOL

To determine the EOL strength of the grids, impact tests were performed using **AP1000** mid grids with gaps between the grid springs and the fuel rods as described in the response to RAI 25 (Reference 3). Gaps that form between the grid springs and the fuel rods are the primary contributor to the reduction in grid strength that occurs at EOL conditions. Gap formation is due to irradiation induced spring relaxation, grid growth, and cladding creep-down. Except for the gaps, these tests were performed using the standard Westinghouse grid impact testing methodology. This is the same methodology that was used for the original BOL **AP1000** grid impact tests.

The average gap used in the **AP1000** EOL impact tests conservatively exceeds the upper [ ]^{a,c} confidence level of the mean gap based on post-irradiation exam (PIE)

measurements of grid cell size and fuel rod diameter as shown in Figure 1 below. These PIE measurements are from fuel assemblies with burn-ups comparable to the **AP1000** EOL burn-ups and with fuel rods with the same diameter and material as the **AP1000** fuel rods. These fuel assemblies have RFA style grids. The RFA design is the basis for the **AP1000** mid grid design. The RFA grids have the same material (ZIRLO®) and strap thicknesses as the **AP1000** grids. [

]^{a,c}

a,c

[

]

Figure 1. Grid Spring to Fuel Rod Gaps at EOL Conditions from PIE Measurements

[

]^{a,c}

The general grid impact test set-up includes a swinging hammer mounted on a four bar linkage, supporting back plates, load cells, angular transducer, furnace, thermocouples, and mounting fixtures. Two different supporting backing plates are used – one to represent impact between grids, and one to represent impact between the grid and the core barrel shroud.

The hammer is released from various increasing angular displacements resulting in increasing grid impact loads until failure of the grid occurs. Failure is characterized as a reduction in the impact load during testing. [

]^{a,c} A picture of a typical failed EOL grid is shown in Figure 2. [

]^{a,c}

a,c

[ ]

Figure 2. Typical EOL Grid Failure Mode

The grid strength is defined as the lower 95% confidence level of the mean failure load. A summary of the results is provided in Table 1. [ ]^{a,c}

Table 1. Summary of Grid Impact Test Results

a,c

[ ]

These results are based on impact tests of the mid grid which is the grid with the highest seismic loading and lowest margins. Additional impact testing of the **AP1000** IFM grids at EOL conditions is not necessary because the IFM grids have a [ ]^{a,c}. As such, the difference between the BOL and the EOL grid strength is expected to be minimal. Furthermore, the strength of the IFM grid is significantly greater than the maximum IFM grid impact load providing sufficient margin [ ]^{a,c} to accommodate any minor EOL effects.

3.0 FUEL ASSEMBLY MODAL FREQUENCIES AT EOL

The **AP1000** fuel assembly described in the Core Reference Report (Reference 1) was tested to determine the assembly EOL modal frequencies. The EOL modal frequencies are lower than the BOL frequencies and result in higher grid impact loads. These frequencies are used to develop the core dynamic analysis models. To simulate EOL conditions the assembly was tested with gapped grid cells; i.e., a clearance between the grid springs and the fuel rods. The mid grid gaps were similar to the gaps used in the EOL grid impact tests. Except for the gaps, these tests were performed using the standard Westinghouse fuel assembly vibration test methodology, which is the same methodology used for BOL tests. Results from the EOL vibration test comparing the BOL and EOL fuel assembly frequencies is summarized Table 2. Based on these test results the first mode frequency at EOL is [ ]^{a,c}.

Table 2. Fuel Assembly Modal Frequencies a,c

[ ]

4.0 FUEL ROD STIFFENING AT EOL

At EOL conditions the fuel rod bending stiffness is increased due to bonding that occurs between the pellet and cladding because of cladding creep down and pellet swelling. Increased fuel rod bending stiffness will offset some of the reduction in the fuel assembly lateral vibration frequencies due to gap formation between the grid springs and the fuel rods. Reduced fuel assembly frequencies result in higher seismic grid impact loads.

[ ]^{a,c}

[

]a,c

a,c

Figure 3. Irradiated Fuel Rod Four Point Bend Test

5.0 FUEL ASSEMBLY DAMPING

The BOL core dynamic analysis was performed using a conservative fuel assembly damping ratio of [ ]a,c. To maintain acceptable grid impact load margins at EOL conditions, higher (more realistic) damping must be considered in the EOL analysis. Justification for higher damping was provided in the responses to RAIs 24 & 27 (References 3 & 4) based on considerations of flowing water effects as discussed below.

Figures 4 and 5 summarize Westinghouse fuel assembly damping test data in air, still water and flowing water for water temperatures between 80°F and 300°F. The test data provides a direct comparison of the damping values due to the different media. These tests are described in Reference 5. An analysis of this data leads to the following observations of the trends associated with damping in flowing water:

1) Damping in flowing water increases slightly with vibration amplitude.
2) Damping in flowing water decreases slightly with increasing temperature.
3) Flow velocity has a strong effect on damping and results in a significant increase in damping with increasing flow velocity.

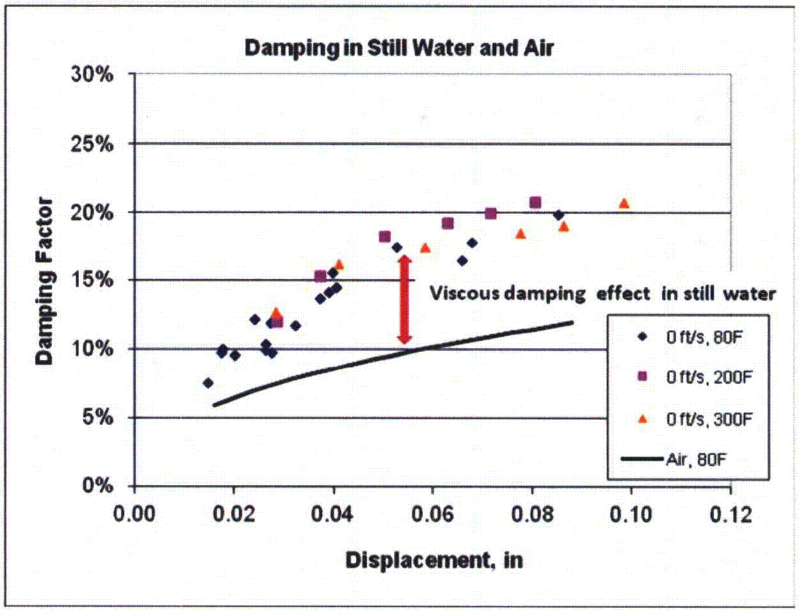

Figure 4. Fuel Assembly Damping Factors in Air and Still Water – Westinghouse Test Data

a,c

Figure 5. Fuel Assembly Damping in Flowing and Still Water – Westinghouse Test Data

These observations are consistent with other test results that are available publicly (References 6 and 7). The considerations for application of hydraulic damping coefficients due to flowing water are discussed further below.

Fuel assembly damping force in flowing water is actually the summation of fuel structural damping in air (due to material and friction damping), viscous damping in still water and hydraulic damping in flowing water as shown in Equation (1). All three damping coefficients are non-linear.

$$F_d = c_s\dot{x} + c_v\dot{x} + c_h\dot{x} \qquad (1)$$

c_s – Structural damping coefficient in air, mainly increasing with amplitude
c_v – Viscous damping in still water, mainly increasing with vibration velocity
c_h – Hydraulic damping in flowing water, mainly increasing with axial flow velocity

Flow Rate Dependence

Both structural damping and viscous damping terms are vibration-amplitude dependent and increase with vibration amplitude. The hydraulic damping term in flowing water is flow rate dependent and it dominates the total damping. Typical fuel assembly displacements during a seismic event with grid impact occurrences are much greater than [ ]^{a,c}. Therefore, the damping data from the flowing water tests with vibration displacements greater than [ ]^{a,c} (see green box in Figure 5) are used to obtain the best fit curve of damping versus flow velocity shown in Figure 6.

a,c

[ ]

Figure 6. Fuel Assembly Damping in Flowing Water

Temperature, Vibration Amplitude, and Measurement Variability Effects

The data variability shown in Figure 6 is due to variations in amplitude, temperature [ ]^{a,c} and measurement. This data is bounded [ ]^{a,c} as shown in Figure 7. The lower bound curve is approximately [ ]^{a,c} below the best fit curve and conservatively covers measurement variability and temperature effects up to [ ]^{a,c}. The lower bound curve is conservative because the

confidence interval was determined based on data variability that includes variations in amplitude between [ ]^{a,c}. Typical fuel assembly displacements during a seismic event with grid impact occurrences are much greater than [ ]^{a,c}. If only higher amplitude data is considered, the best fit curve would be higher and the lower bound curve would be closer to the best fit curve. It is observed that the lower bound curve covers all test data points except one at high flow rate, [ ]^{a,c}.

a,c

Figure 7. Statistical Evaluation of Damping Data

As shown in Figure 5, damping is not very sensitive to temperature. For example, [

]^{a,c}. This is consistent with data from Reference 7 where it was concluded that in the range between 70° to 600°F "damping is minimally affected by temperature in water." To account for temperatures up to [ ]^{a,c}, the lower bound damping curve is conservatively reduced by an additional [ ]^{a,c} as shown in Figure 8.

a,c

Figure 8. Damping Design Curve

The design damping curve is defined [ ]^{a,c} below the best fit curve. This conservatively envelops the combined effects of measurement uncertainty and temperature up to [ ]^{a,c}. It should be noted that a [ ]^{a,c} reduction in the damping coefficient corresponds to approximately a [ ]^{a,c} reduction in the damping coefficient [ ]^{a,c}.

The damping in flowing water is the summation of structural damping, viscous damping and hydraulic damping as shown in Equation (1). The damping curves in Figure 8 show the tendency of damping to increase with flow rate. Based on Equation (1), the damping at 0 ft/s should be equal to the damping in still water; however, extrapolation of the damping design curve to a flow velocity of 0 ft/s results in a damping coefficient of [ ]^{a,c}. As such, the design damping curve is conservative, because the damping coefficient in still water (flow velocity = 0 ft/s) is [ ]^{a,c}.

Effect of Design Uncertainty on Damping in Flowing Water

The fuel assembly used in the Westinghouse flowing water damping tests was a PWR fuel assembly [ ]^{a,c}. Although the array size, number of thimbles, and number of mid grids vary among fuel assembly designs, the basic structure of this test assembly is similar to all other PWR fuel assemblies including the **AP1000** fuel assembly. The Westinghouse test data provides a direct comparison and clearly shows the differences of damping values due to the different media (air, still water, and flowing water).

Other fuel vendors have also performed fuel assembly damping tests with similar but not identical PWR fuel assemblies to the Westinghouse test fuel assembly. The test assembly from Reference 6 has a 17x17 array, 8 mid grids and 264 fuel rods with 0.374 inch OD. These parameters are the same as for the **AP1000** fuel assembly. Reference 6 provides similar damping values and demonstrates similar damping characteristics such as (1) the damping values in air and still water are amplitude dependant and (2) the damping in flowing water is significantly higher than in air and still water and is less amplitude dependant. The test assembly from Reference 7 also has a 17x17 array and 264 fuel rods with 0.374 inch OD and also provides similar damping values. The main reason why fuel assembly damping coefficients obtained from tests of different designs are similar is because of the geometric similarity of the various designs. All of these fuel assemblies are comprised of a square array of fuel rods with guide thimble tubes and spacer grids. For this reason Westinghouse currently uses the same damping coefficient for all types of Westinghouse fuel in Westinghouse type reactors.

Comparisons of the deltas between in-air and flowing water damping among various fuel assembly designs are provided in Table 3. This data shows that the deltas among the designs are very similar. By comparing the deltas, the uncertainties from the various tests are minimized and the effects of flowing water are emphasized.

The deltas between in-air and flowing water damping, shown in Table 3, exclude the fuel assembly mechanical damping in air. By adding a conservative mechanical damping of

[ ]a,c to any of the damping values in Table 3, the damping in flowing water will be greater than [ ]a,c, which is the damping from the design curve at [ ]a,c. It should be noted that the **AP1000** fuel assembly mechanical damping in air at EOL conditions is [ ]a,c. Therefore, the damping design curve is conservative and no additional reductions are necessary to account for differences between the tested fuel assembly design and the **AP1000** fuel assembly design.

Table 3*. Delta between Flowing Water (FW) and In-Air Damping a,c

[ ]

Pump Coastdown

[

]a,c

a,c

[

]

Figure 9. RCS Pump Coastdown

The damping coefficient as a function of time during a pump coastdown is shown in Figure 10. This Figure was generated using the damping design curve from Figure 8 and the pump coastdown curve from Figure 9 assuming the Technical Specification minimum flow at the start of the coastdown. [ ]^{a,c} This is the damping value that is used in the core dynamic seismic analysis.

a,c

[

]

Figure 10. Damping during RCP Coastdown

It should be noted that damping is a cumulative effect for dissipating vibration energy over time. [ ]^{a,c} As such, using [ ]^{a,c} damping in the seismic analysis is conservative.

6.0 EOL SEISMIC/LOCA ANALYSIS

The BOL core dynamic analysis models were updated based on data from the EOL grid impact tests and fuel assembly mechanical tests. These updated models were used to perform the EOL fuel assembly seismic/LOCA analysis. A damping value of [ ]^{a,c} was used in the seismic analysis. The LOCA analysis was conservatively performed assuming [ ]^{a,c} damping.

The dominant grid impact loads occur during a seismic event. The LOCA grid impact loads are negligible. This is because the main reactor coolant pipes in an **AP1000** plant are qualified for leak-before-break. Results from the analysis indicate that the limiting grid impact load is [ ]^{a,c}. The EOL grid strength is [ ]^{a,c} resulting in a margin of [ ]^{a,c}. Therefore, requirements for control rod insertability are met.

These results are conservative. They are based on a generic bounding seismic spectrum considering various soil conditions. The site specific seismic spectrum at any individual **AP1000** site is expected to be less limiting. In addition, the fuel assembly dynamic models don't consider [ ]^{a,c} If these effects were included, it is expected that grid impact loads would be less.

To insure fuel rod fragmentation does not occur, additional EOL seismic cases were run with [ ]^{a,c} damping. Results from these cases demonstrate that the limiting fuel rod stress [ ]^{a,c} remains significantly below the allowable limit [ ]^{a,c}.

Analysis of operating basis earthquake (OBE) loads is not required according to the DCD (Tier 2, Section 3.7 of Reference 8); however, to insure that no grid deformation occurs during an OBE, an evaluation of OBE loads was performed. Typically, OBE analyses are performed assuming the plant is operating at full power conditions; however, in this case the evaluation was conservatively performed assuming no flow [ ]^{a,c}. The OBE is one-third of the SSE in accordance with the DCD (Tier 2, Section 3.7 of Reference 8). [ ]^{a,c} As such, no grid deformation is expected during an OBE.

7.0 SUMMARY AND CONCLUSIONS

Results demonstrate that the limiting combined seismic and LOCA grid impact loads are less than the grid strength at EOL conditions [ ]^{a,c}.

The EOL strength of the grids was determined from impact tests performed on **AP1000** grids with gaps between the grid springs and the fuel rods. Gap formation between the grid springs and the fuel rods due to irradiation induced spring relaxation, grid growth, and cladding creep-down is the primary contributor to the reduction in grid strength that occurs at EOL conditions.

The grid impact loads were determined from core dynamic analyses. The analytical models in these analyses were adjusted for EOL effects based on properties obtained from the EOL grid impact tests and the EOL fuel assembly mechanical test. To simulate EOL conditions, the fuel assembly mechanical test was performed with gapped grid cells; i.e., a clearance between the grid springs and the fuel rods.

The seismic loads were calculated based on a fuel assembly damping coefficient of [ ]^{a,c}. This damping coefficient is based on Westinghouse test data considering hydraulic damping due to flowing water effects. The flow velocity used to determine the damping coefficient conservatively assumed that the RCS pumps would trip at the start of the seismic event and that the flow rate at the beginning of the pump coastdown is equal to the Technical Specification minimum flow. The Westinghouse damping data is consistent with test data reported by others. The Westinghouse data was conservatively adjusted for temperature effects and measurement uncertainty. The effect of design differences between the tested fuel assembly and the **AP1000** fuel assembly were judged to be minor based on a review of test data from various designs.

REFERENCES

1. Hone, M., et al., "**AP1000** Core Reference Report," WCAP-17524-P (Proprietary) and WCAP-17524-NP (Non-Proprietary), March 2012, Westinghouse Electric Company LLC.

2. Tonacci, M., "Acceptance for Review of Westinghouse Topical Report WCAP-17524-P, Revision 0, **AP1000** Core Reference Report," ML12144A201, June 5, 2012, U.S. Nuclear Regulatory Commission.

3. Gresham, J.A., "Westinghouse Response to NRC RAIs on WCAP-17524, **AP1000** Core Reference Report (Proprietary/Non-Proprietary)," LTR-NRC-12-86, January 2, 2013, Westinghouse Electric Company LLC.

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7. F. E. Stokes and R. A. King, "PWR Fuel Assembly Dynamic Characteristics," International Conference on Vibration in Nuclear Power Plants, Keswick, United Kingdom, May 9-12, 1978 (BNES).

8. Westinghouse **AP1000** Design Control Document, Rev 19, ML11171A500, June 2011.

Section I

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LTR-NRC-13-43

June 26, 2013

Subject: Supplemental Information Related to Closure of the Corrective Action Identified in the Response to CRR-001 (Proprietary/Non-Proprietary)

Enclosed are copies of the proprietary and non-proprietary versions of supplemental information related to the closure of the corrective action identified in the response to CRR-001 for the **AP1000®** Pressurized Water Reactor (PWR) Core Reference Report (CRR) (WCAP-17524-P). This letter is being transmitted to address the commitment identified in the response to CRR-001 transmitted in LTR-NRC-13-3.

Also enclosed is:

1. One (1) copy of the Application for Withholding Proprietary Information from Public Disclosure, AW-13-3736 (Non-Proprietary), with Proprietary Information Notice and Copyright Notice.
2. One (1) copy of Affidavit (Non-Proprietary).

This submittal contains proprietary information of Westinghouse Electric Company LLC. In conformance with the requirements of 10 CFR Section 2.390, as amended, of the Commission's regulations, we are enclosing with this submittal an Application for Withholding Proprietary Information from Public Disclosure and an affidavit. The affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference AW-13-3736 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 310, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

James A. Gresham, Manager
Regulatory Compliance

Enclosures

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AW-13-3736

June 26, 2013

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-NRC-13-43 P-Attachment, "Supplemental Information Related to Closure of the Corrective Action Identified in the Response to CRR-001" (Proprietary)

Reference: Letter from James A. Gresham to Document Control Desk, LTR-NRC-13-43, dated June 26, 2013

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC (Westinghouse), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.390, Affidavit AW-13-3736 accompanies this Application for Withholding Proprietary Information from Public Disclosure, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject 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 proprietary aspects of the application for withholding or the accompanying affidavit should reference AW-13-3736 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 310, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

J. A. Gresham

James A. Gresham, Manager
Regulatory Compliance

Enclosures

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared James 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

James A. Gresham, Manager
Regulatory Compliance

Sworn to and subscribed before me
this 26th day of June 2013

Anne M. Stegman

Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Anne M. Stegman, Notary Public
Unity Twp., Westmoreland County
My Commission Expires Aug. 7, 2016
MEMBER, PENNSYLVANIA ASSOCIATION OF NOTARIES

(1) I am Manager, Regulatory Compliance, 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 Proprietary Information from Public Disclosure 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-NRC-13-43 P-Attachment, "Supplemental Information Related to Closure of the Corrective Action Identified in the Response to CRR-001" (Proprietary), for submittal to the Commission, being transmitted by Westinghouse letter, LTR-NRC-13-43, 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 review of WCAP-17524, and may be used only for that purpose.

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

(a) Obtain NRC approval of the **AP1000**® Pressurized Water Reactor (PWR) Advanced First Core, as documented in WCAP-17524, "AP1000 Core Reference Report".

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of the information to its customers for the purpose of assisting customers in obtaining license changes for the **AP1000** PWR.

(b) This document establishes a portion of the licensing basis for the **AP1000** PWR.

(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 technical evaluation justifications 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.

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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).

Supplemental Information Related to Closure of the Corrective Action Identified in the Response to CRR-001 (Non-Proprietary)

June 2013

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Cranberry Township, PA 16066

Supplemental Information Related to Closure of the Corrective Action Identified in the Response to CRR-001

As part of the response provided to CRR-001 on the Core Reference Report (WCAP-17524-P) that was contained in Westinghouse letter LTR-NRC-13-3, Westinghouse identified a minor discrepancy in the modeling of the diameter of the GRCA absorber. Westinghouse has completed an assessment of the discrepancy and subsequently closed the corrective action. Results of the analysis show that the maximum impact over the entire burnup range is [ ]^{a,c}. This is small compared to the [ ]^{a,c} reserved in the analysis and has no impact on the conclusions or results of the analysis. Therefore, the spent fuel storage racks continue to meet the regulatory requirements contained in 10CFR 50.68.

Section J

Westinghouse

Westinghouse Electric Company
Engineering, Equipment and Major Projects
1000 Westinghouse Drive
Cranberry Township, Pennsylvania 16066
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

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Direct fax: (724) 720-0754
e-mail: greshaja@westinghouse.com

LTR-NRC-13-81

December 10, 2013

Subject: Submittal of Presentations from the NRC Audit of Calculations Supporting WCAP-17524, "AP1000 Core Reference Report" (Proprietary/Non-Proprietary)

Enclosed are the proprietary and non-proprietary versions of the three presentations, "AP1000® Plant SBLOCA CRR Audit Response For Containment Iteration," "SBLOCA CRR Containment Minimum Backpressure Calculation Audit Reponses for Initial Air Temperature," and "SBLOCA CRR Containment Minimum Backpressure Calculation Audit Response for Impact of IOZ Thermal Conductivity" which were presented at the meeting held at the Westinghouse offices on November 25, 2013 and November 26, 2013. This meeting was in support of WCAP-17524, "AP1000 Core Reference Report."

Also enclosed is:

1. One (1) copy of the Application for Withholding Proprietary Information from Public Disclosure, AW-13-3865 (Non-Proprietary), with Proprietary Information Notice and Copyright Notice.
2. One (1) copy of Affidavit (Non-Proprietary).

This submittal contains proprietary information of Westinghouse Electric Company LLC. In conformance with the requirements of 10 CFR Section 2.390, as amended, of the Commission's regulations, we are enclosing with this submittal an Application for Withholding Proprietary Information from Public Disclosure and an Affidavit. The Affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse Affidavit should reference AW-13-3865, and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 310, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

J A Gresham

James A. Gresham, Manager
Regulatory Compliance

Enclosures

Westinghouse

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AW-13-3865

December 10, 2013

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-NRC-13-81 P-Attachment, "Submittal of Presentations from the NRC Audit of Calculations Supporting WCAP-17524, 'AP1000 Core Reference Report'" (Proprietary)

Reference: Letter from James A. Gresham to Document Control Desk, LTR-NRC-13-81, dated December 10, 2013

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC (Westinghouse), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.390, Affidavit AW-13-3865 accompanies this Application for Withholding Proprietary Information from Public Disclosure, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject 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 proprietary aspects of the application for withholding or the accompanying Affidavit should reference AW-13-3865, and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 310, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

J A Gresham

James A. Gresham, Manager
Regulatory Compliance

Enclosures

AW-13-3865

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared James 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

James A. Gresham, Manager

Regulatory Compliance

Sworn to and subscribed before me

this 10th day of December 2013

Anne M. Stegman

Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Anne M. Stegman, Notary Public
Unity Twp., Westmoreland County
My Commission Expires Aug. 7, 2016
MEMBER, PENNSYLVANIA ASSOCIATION OF NOTARIES

(1) I am Manager, Regulatory Compliance, in Engineering, Equipment and Major Projects, 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 Proprietary Information from Public Disclosure 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.

(iii) 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.

(iv) 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.

(v) 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.

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in LTR-NRC-13-81 P-Attachment, "Submittal of Presentations from the NRC Audit of Calculations Supporting WCAP-17524, 'AP1000 Core Reference Report'" (Proprietary) for a meeting held on November 25, 2013 and November 26, 2013, for submittal to the Commission, being transmitted by Westinghouse letter, LTR-NRC-13-81, 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 review of WCAP-17524, "AP1000 Core Reference Report" and may be used only for that purpose.

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

(i) Obtain NRC approval of the **AP1000**®[1] Pressurized Water Reactor (PWR) Advanced First Core, as documented in WCAP-17524, "AP1000 Core Reference Report."

(b) Further this information has substantial commercial value as follows:

(i) Westinghouse plans to sell the use of the information to its customers for the purpose of assisting customers in obtaining license changes for the **AP1000** PWR.

(ii) This document established a portion of the licensing basis for the **AP1000** PWR.

(iii) 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 technical evaluation justifications 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.

---

[1] **AP1000** is a trademark or registered trademark of Westinghouse Electric Company LLC, its affiliates and/or its subsidiaries in the United States of America and may be registered in other countries throughout the world.

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).

Submittal of Presentations from the NRC Audit of Calculations Supporting WCAP-17524, "AP1000 Core Reference Report" (Non-Proprietary)

December 2013

Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, PA 16066

AP1000® Plant SBLOCA CRR Audit Response For Containment Iteration

Andre F. Gagnon
Principal Engineer
LOCA Integrated Services

Westinghouse

Executive Summary

- Previous meeting held on 9/17/2013 at Westinghouse's Rockville office resulted in this question.
  1. Clarify starting with atmospheric pressure during its iterations of NOTRUMP (for mass and energy release) and GOTHIC (for containment pressure) and to justify their termination criteria. [This is because each iteration would be expected to result in a lower mass and energy release and consequently lower containment pressure.]

Westinghouse

Overview

- Summarize Small Break LOCA (SBLOCA) changes from the Core Reference Report (CRR), Rev. 0
- Present iteration approach for SBLOCA containment response

Westinghouse

Background

- Changes to SBLOCA analysis described in Revision 0 of the Core Reference Report (CRR; WCAP-17524-P)
  - Single failure assumption
    - ADS-4 on non-PRHR side (e.g., ADS 4-2)
  - Enhanced containment backpressure for select breaks
    - Containment backpressure used currently for Double-ended Direct Vessel Injection (DEDVI) line break
    - Transient pressure history now being utilized for 2 inch Cold Leg Break and Inadvertent Automatic Depressurization System (INADS) simulations

Westinghouse

Containment Pressure Response Generation

- Mass and energy (MNE) releases from atmospheric containment pressure SBLOCA analysis generated with NOTRUMP
  - 2 inch Cold Leg Break
  - Inadvertent ADS
- MNE releases utilized in WGOTHIC minimum containment pressure model to generate containment pressure response for associated SBLOCA cases
- Iteration performed with 2 inch Cold Leg Break utilizing variable containment MNE releases to demonstrate effect

Westinghouse

General Philosophy

- SBLOCA containment response only important up to IRWST initiation
  - Reduces predicted injection gap duration
  - Subsequent to IRWST initiation, containment pressure differences are associated with the changes in ADS discharge characteristics due to shortened uncovery period
  - Changes to containment pressure subsequent to IRWST initiation have only a minor effect on the SBLOCA behavior
- Iterations with containment group beyond those performed not needed

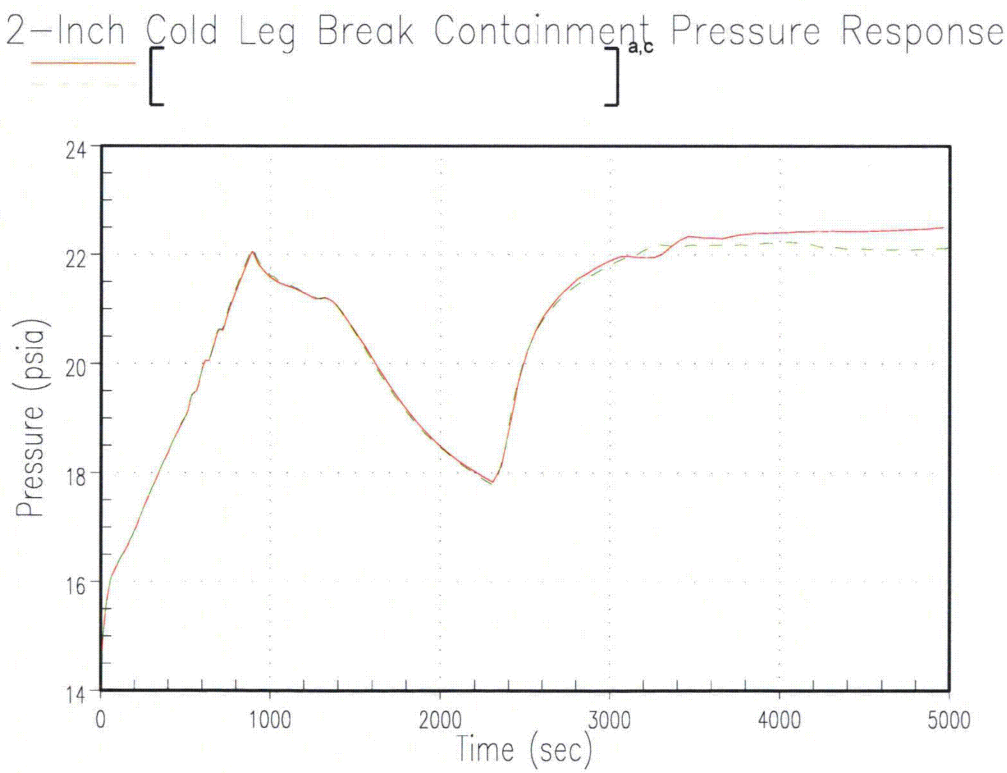
Westinghouse

2 inch SBLOCA Iteration Results

- Initial variable containment pressure response based on previous iteration of NOTRUMP variable containment simulation
  - A comparison between the M&E releases utilized for the generation of the variable containment pressure response and final NOTRUMP simulation was performed
    - No significant changes in simulation response and M&E releases observed
    - Negligible changes in containment pressure expected as a result if they were to be simulated with WGOTHIC
  - No additional iterations required

Westinghouse

Containment Pressure Response (2 inch Cold Leg Break)



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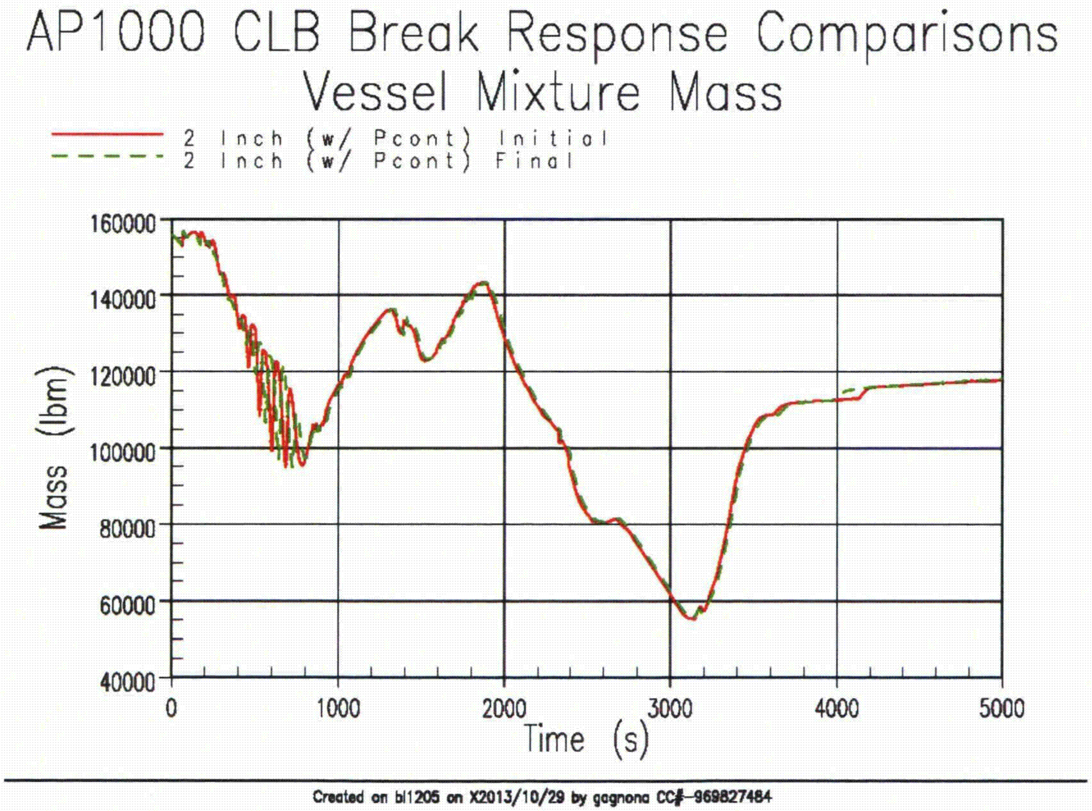
Containment Pressure Response (2 inch Cold Leg Break) Initial vs. Final

a,c

Westinghouse

2 inch SBLOCA Vessel Inventory (Iteration)



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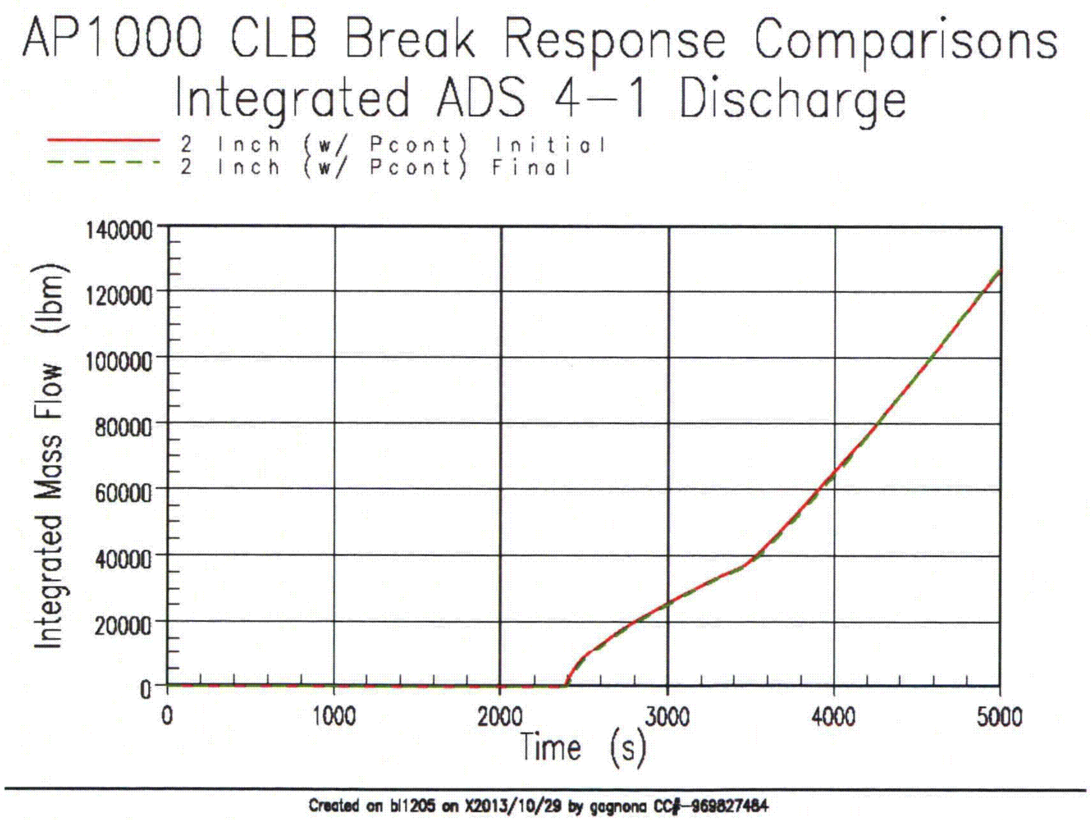
2 inch SBLOCA ADS 4-1 Integrated Discharge (Iteration)



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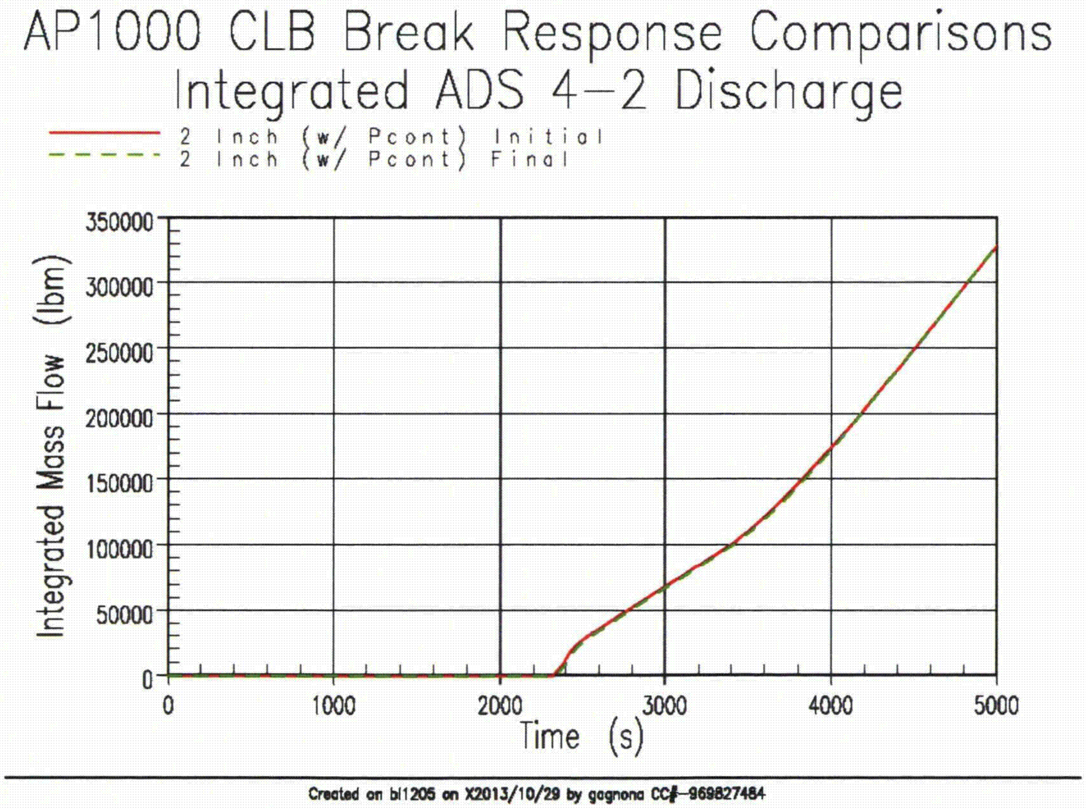
2 inch SBLOCA ADS 4-2 Integrated Discharge (Iteration)



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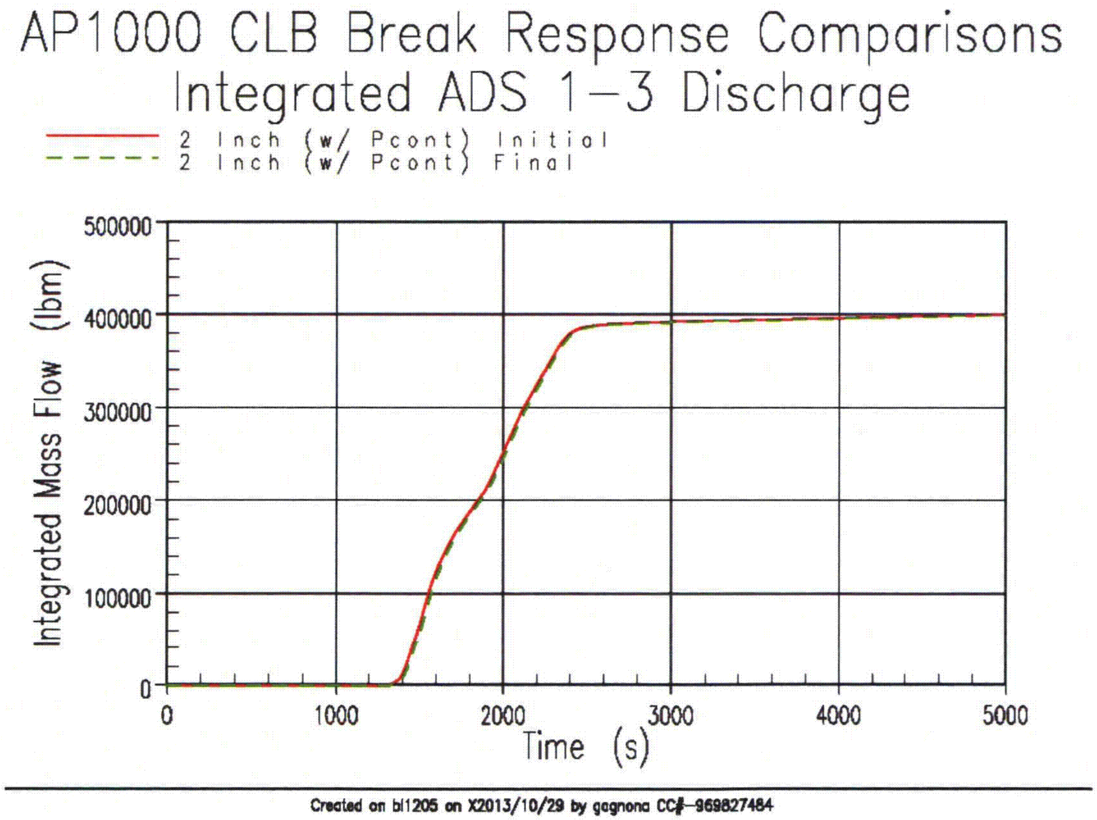
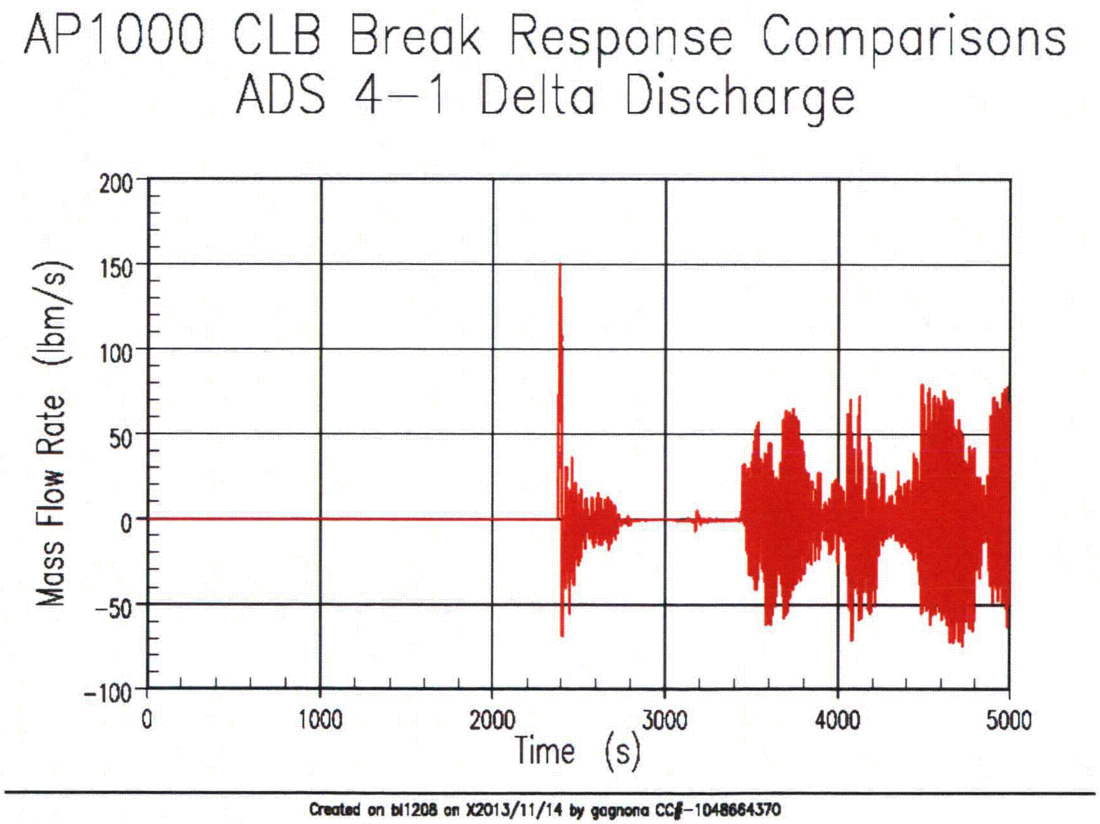
2 inch SBLOCA ADS 1-3 Integrated Discharge (Iteration)



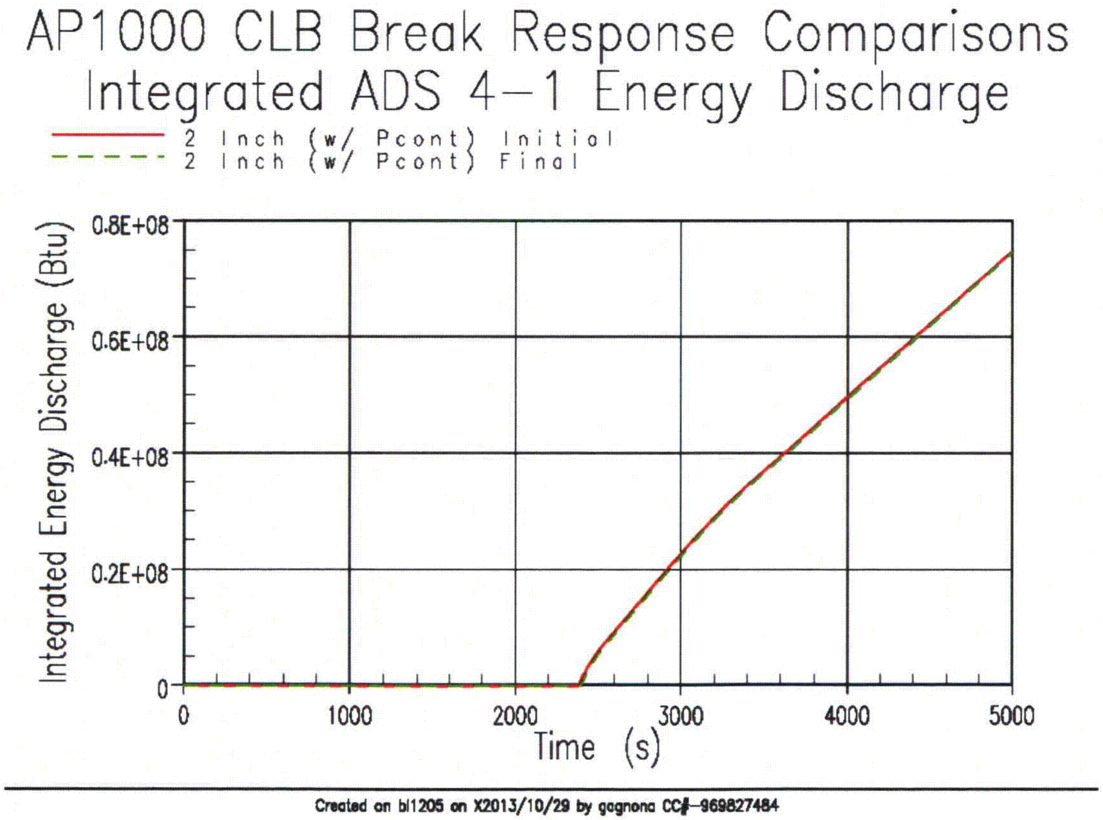
Westinghouse

2 inch SBLOCA ADS 4-1 Discharge Differences (Iteration)



Westinghouse

2 inch SBLOCA ADS 4-1 Integrated Energy Discharge (Iteration)



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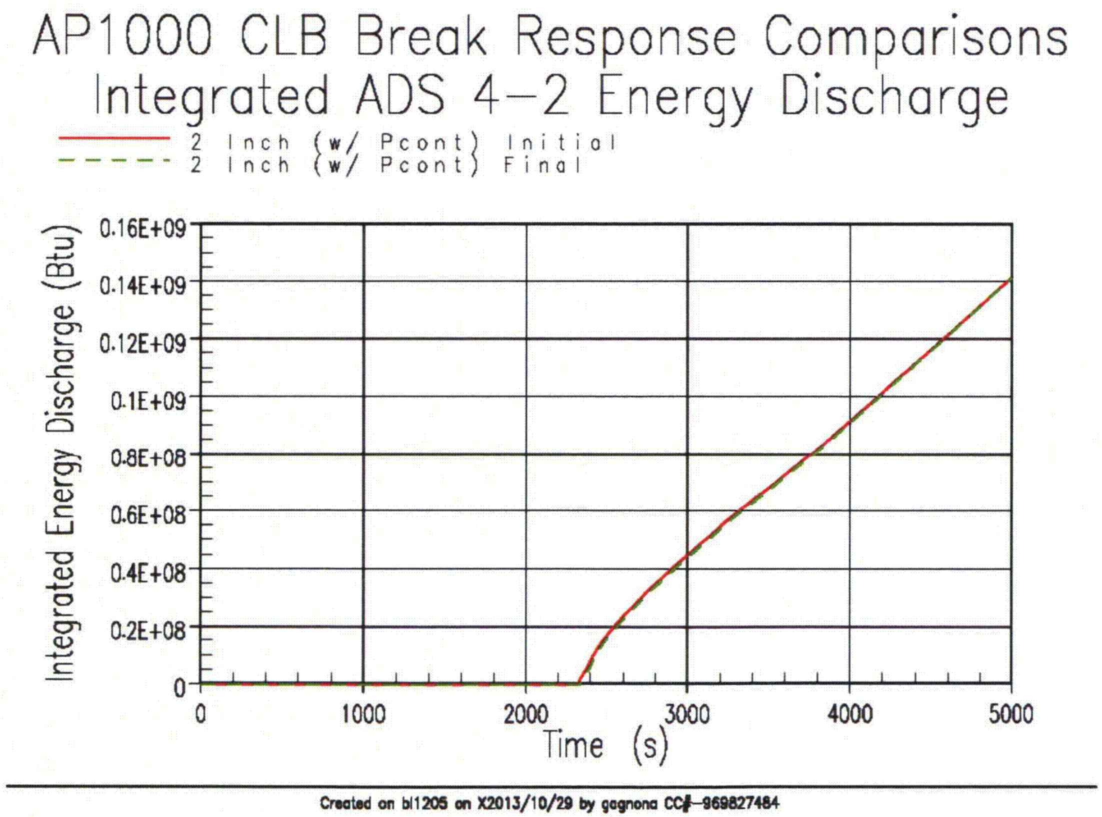
2 inch SBLOCA ADS 4-2 Integrated Energy Discharge (Iteration)



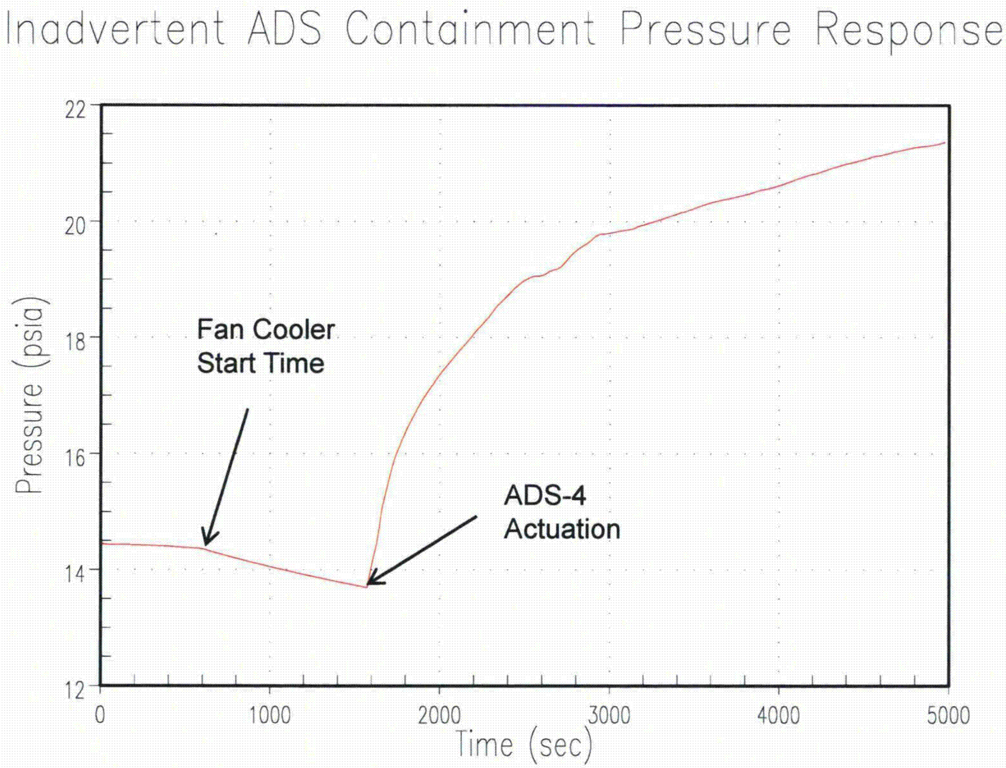
Westinghouse

INADS SBLOCA Iteration Results

- Variable containment pressure response based on Atmospheric NOTRUMP simulation
  - A comparison between the M&E releases utilized between the NOTRUMP simulations was performed
    - Changes in simulation response and M&E releases observed post IRWST injection
    - Changes in containment pressure expected if they were to be simulated with WGOTHIC
  - NOTRUMP sensitivities to containment pressure performed to demonstrate SBLOCA effect
    - Hold Pcont= [ ]^{a,c} psia at ~IRWST Start (2474 sec)
    - Hold Pcont= [ ]^{a,c} psia at MNE divergence time (2921 sec)
    - Results indicate important NOTRUMP results are insensitive to containment pressure following IRWST onset
- Additional iteration performed using M&E releases from NOTRUMP variable containment run to confirm findings
  - Negligible change in responses observed

Westinghouse

Containment Pressure Response (Inadvertent ADS)



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INADS Constant vs. Variable Pressure

a,c

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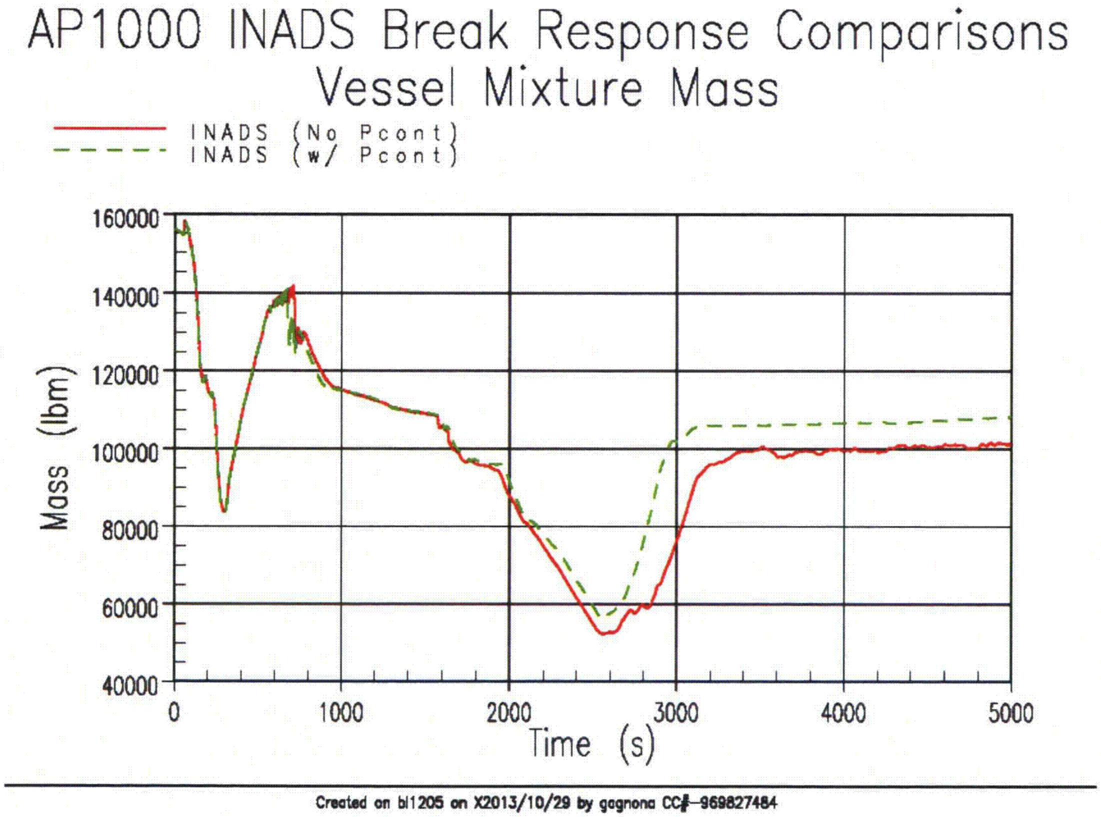
INADS – Vessel Inventory



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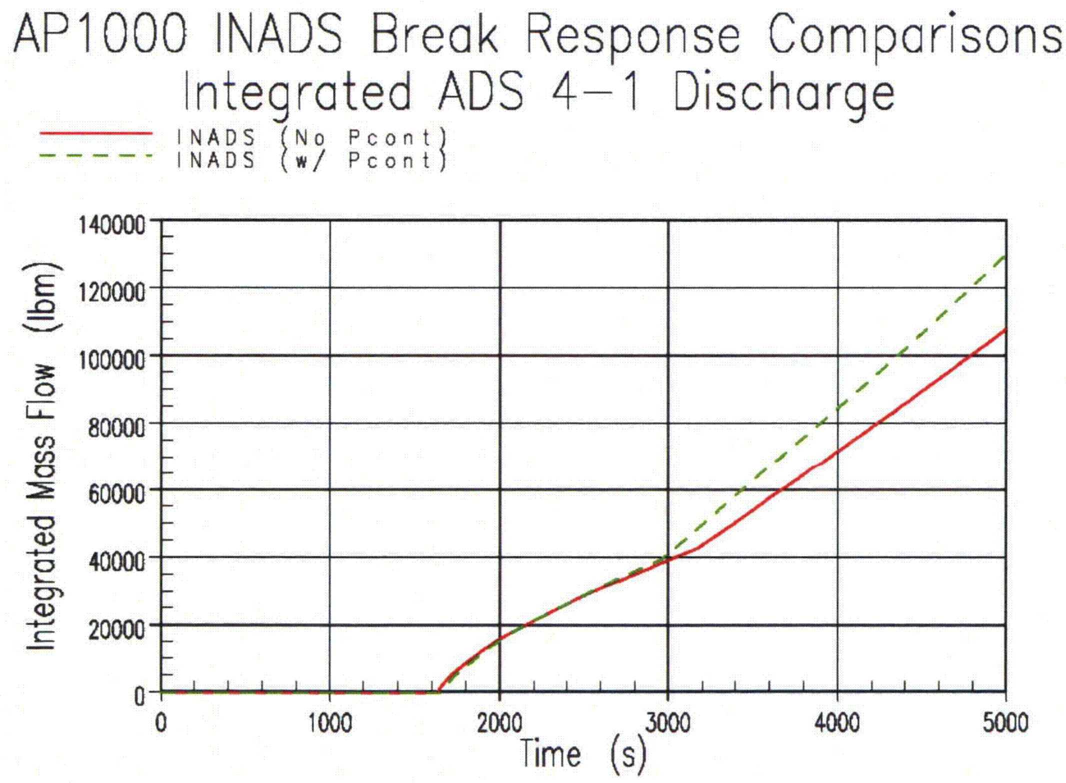
INADS – Integrated ADS 4-1 Discharge



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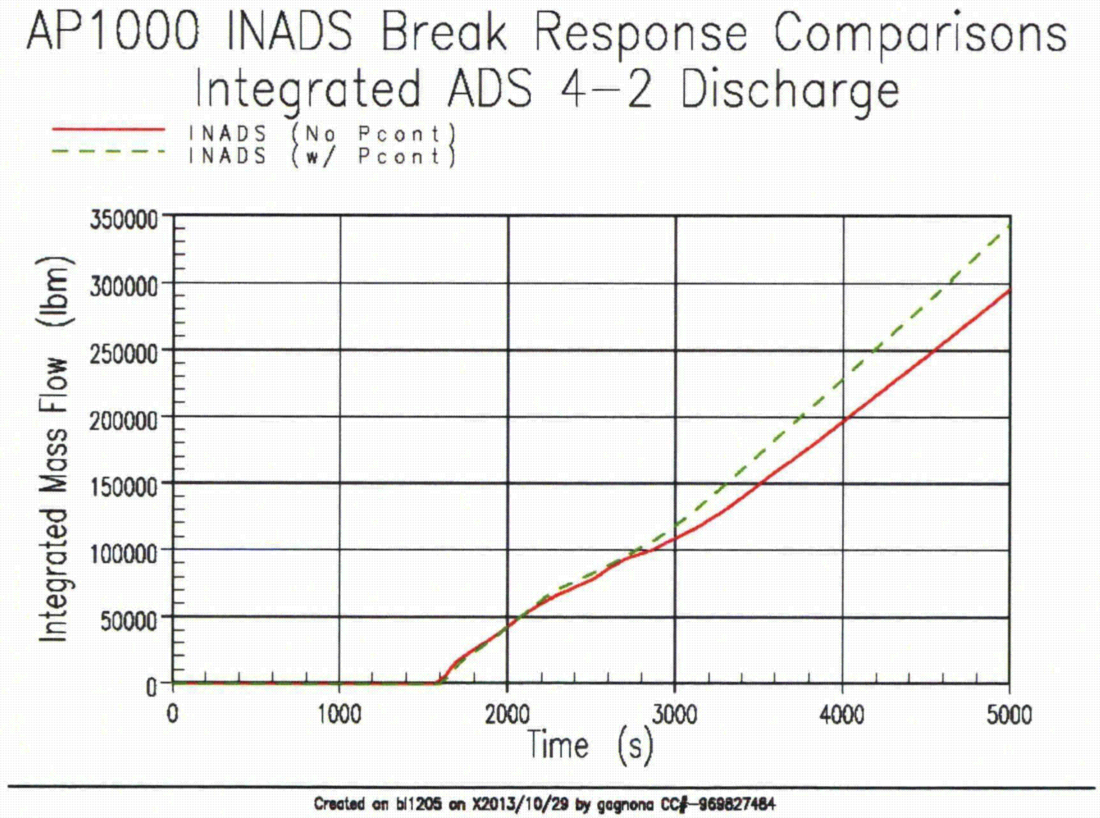
INADS – Integrated ADS 4-2 Discharge



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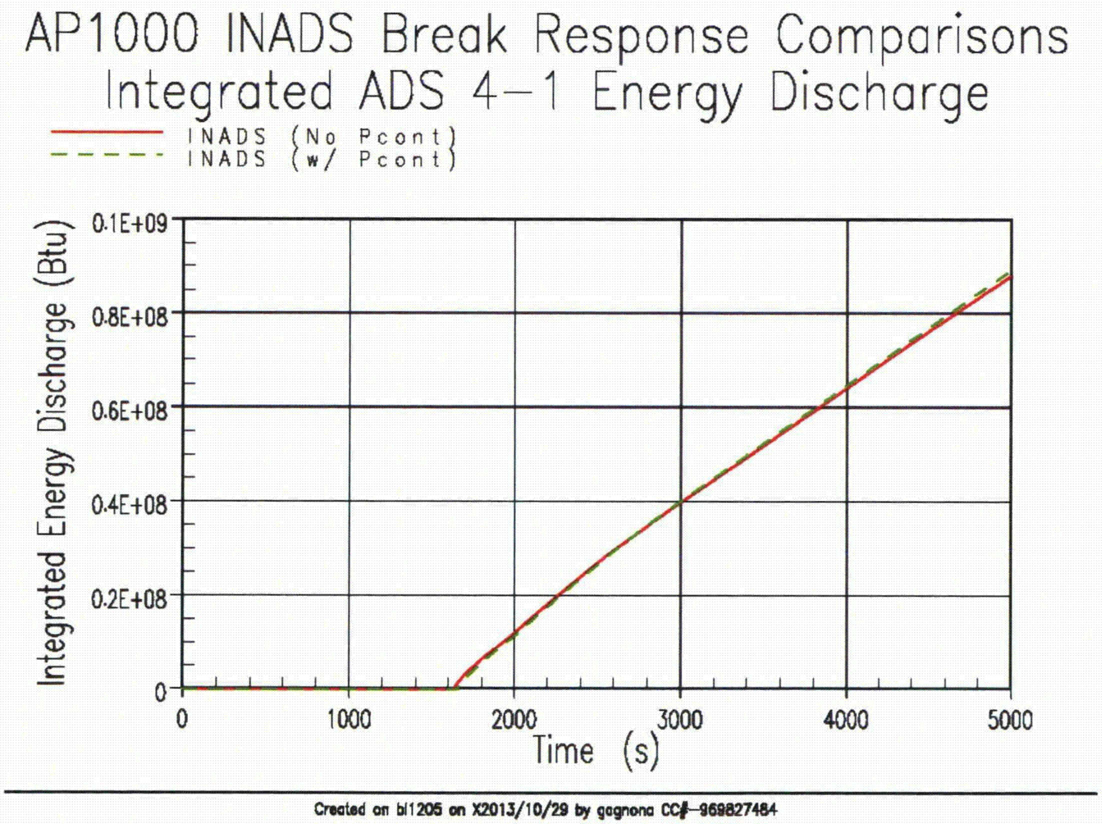
INADS – Integrated ADS 4-1 Energy Discharge



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INADS – Integrated ADS 4-1 Equivalent Discharge Enthalpy

[ ] a,c

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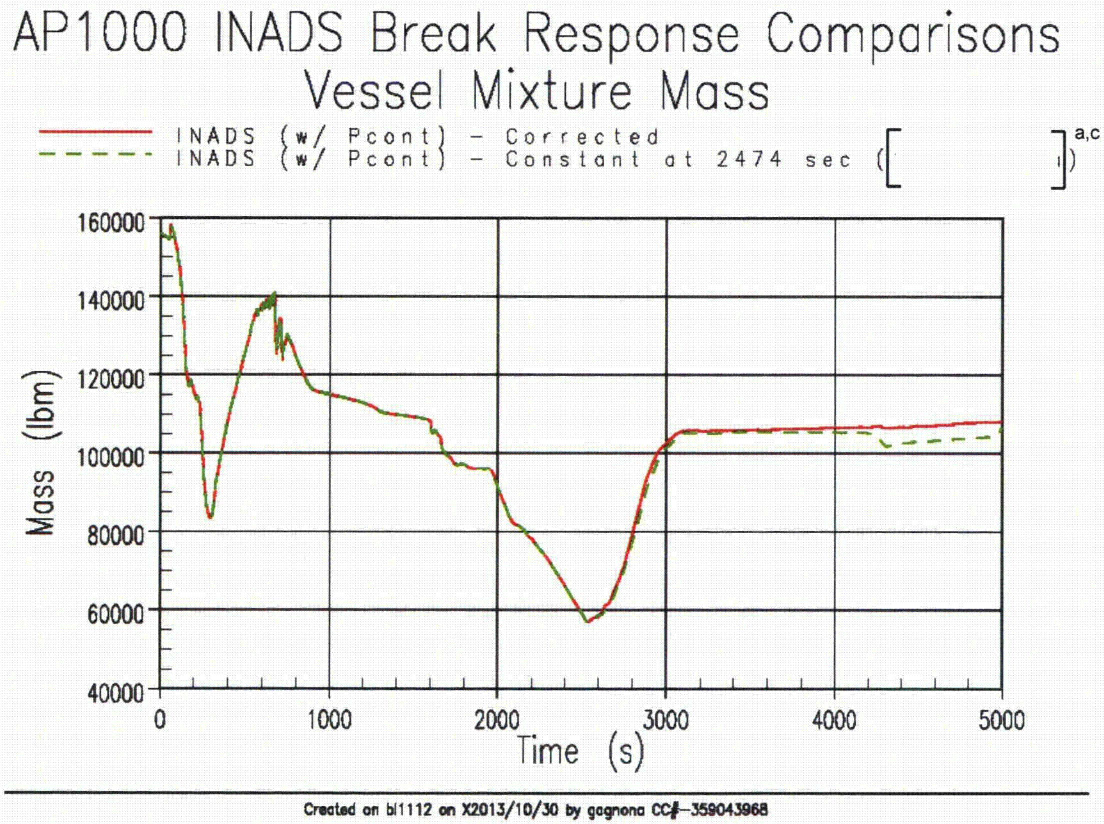
INADS – Clip Containment Pressure At ~IRWST Start

[ ] a,c

INADS – Clip Containment Pressure At ~IRWST Start: Vessel Inventory



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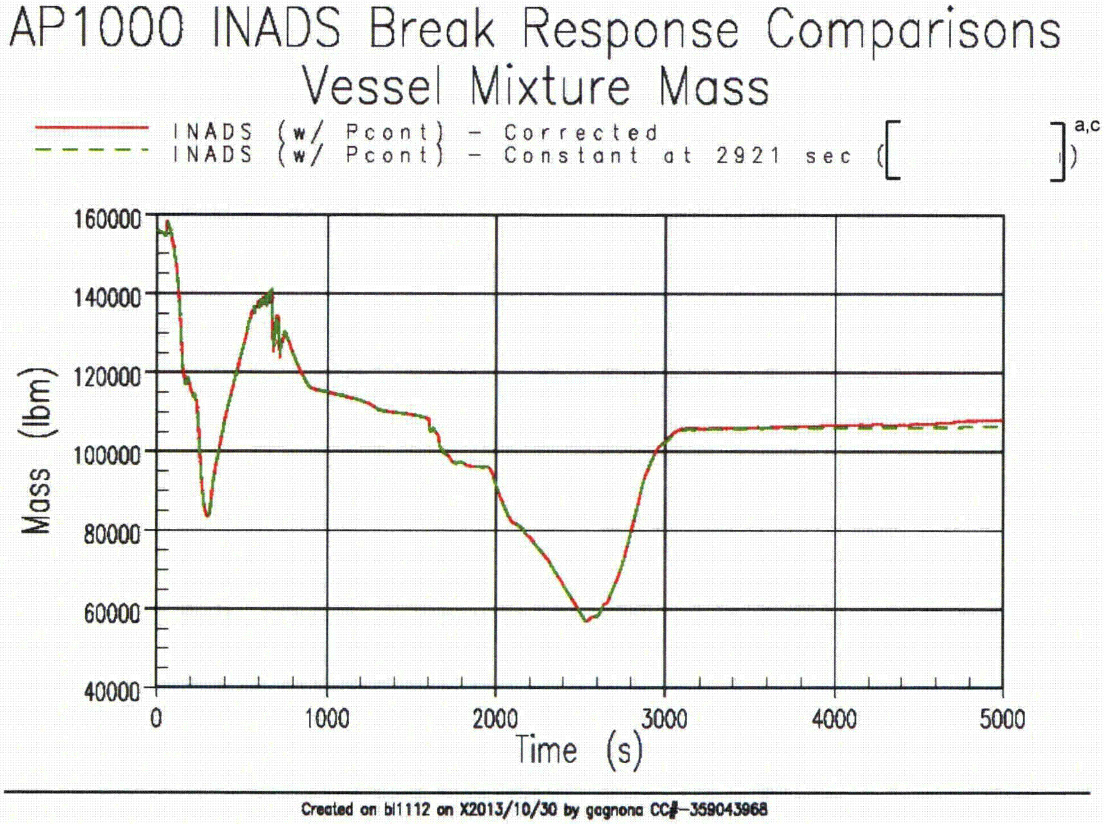
INADS – Clip Containment Pressure At ADS 4-1 Divergence

a,c

Westinghouse

INADS – Clip Containment Pressure At ADS 4-1 Divergence: Vessel Inventory



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INADS Iteration 1

- To further confirm conclusions reached with sensitivities, NOTRUMP M&E releases from variable containment utilized to obtain new containment response
- Utilized new containment response in NOTRUMP simulation
- Compare NOTRUMP MNE releases to those generated by original variable containment run
  - No significant differences observed
  - Some minor timing shifts; however, this is unique to the INADS simulations ([ ]^{a,c} actuation time shifted [ ]^{a,c} seconds) due to initial sub-atmospheric behavior

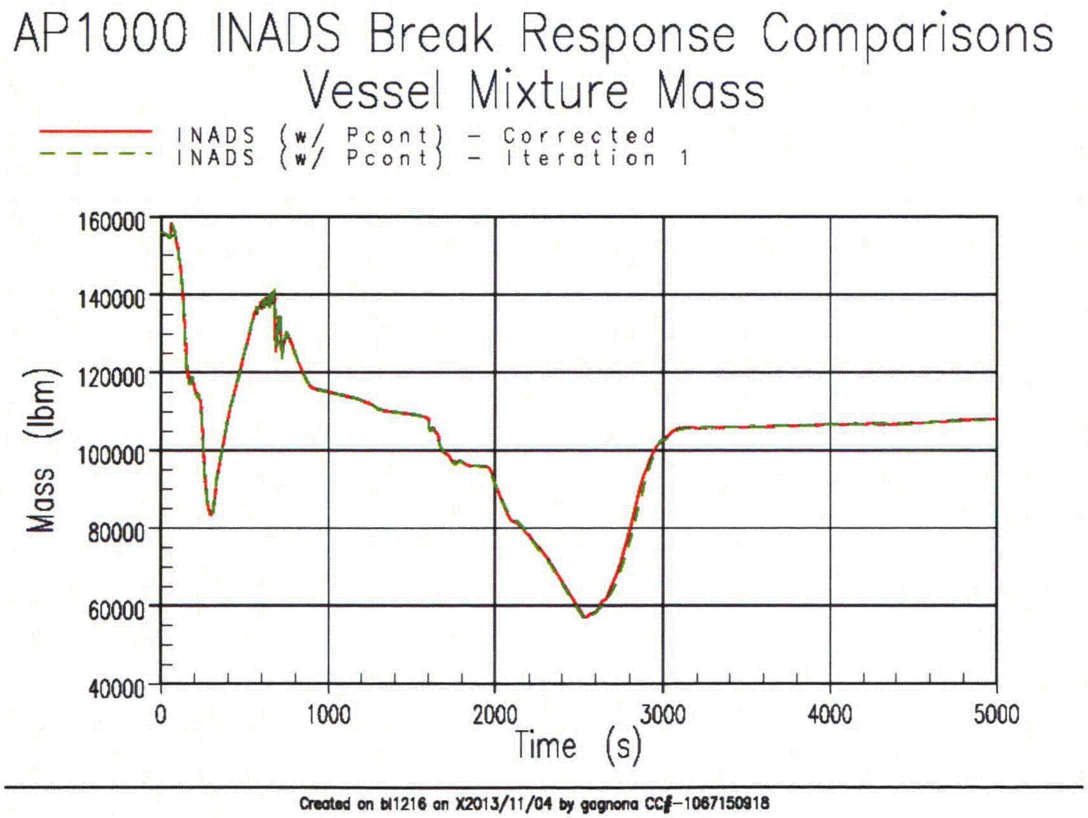
Westinghouse

INADS Iteration 1 – Containment Response

a,c

Westinghouse

INADS Iteration 1 – Vessel Inventory



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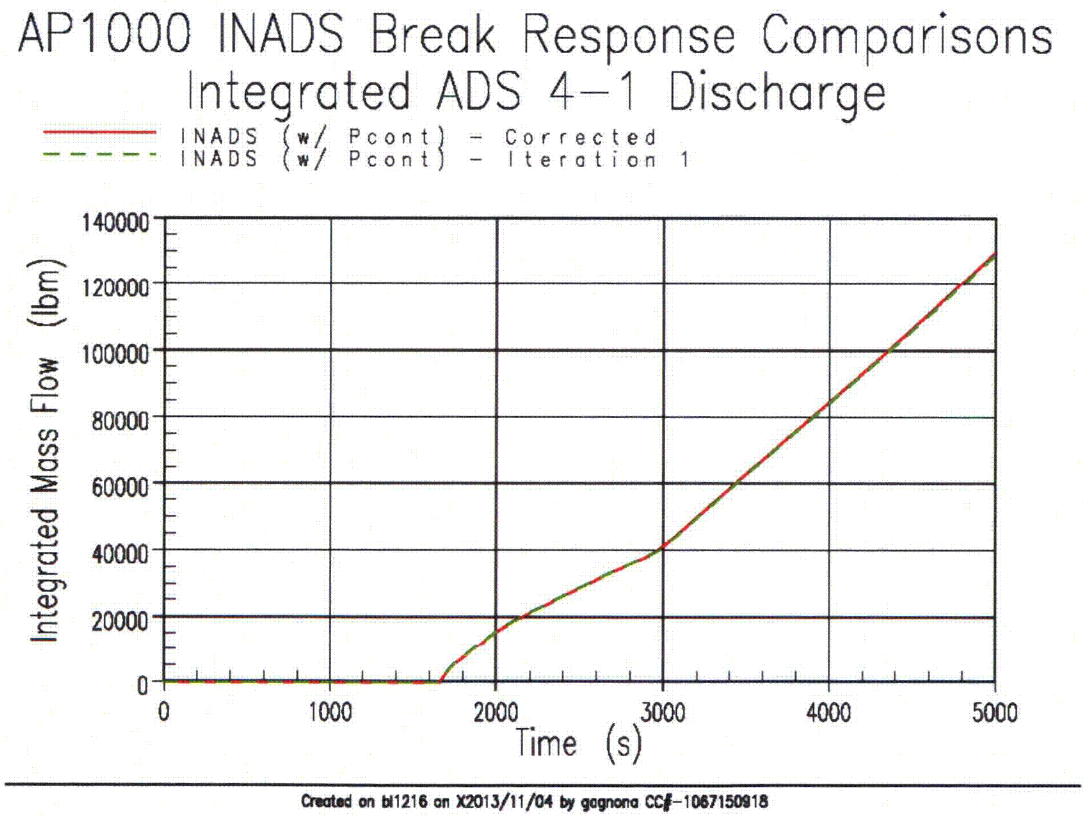
INADS Iteration 1 – Integrate ADS 4-1 Discharge



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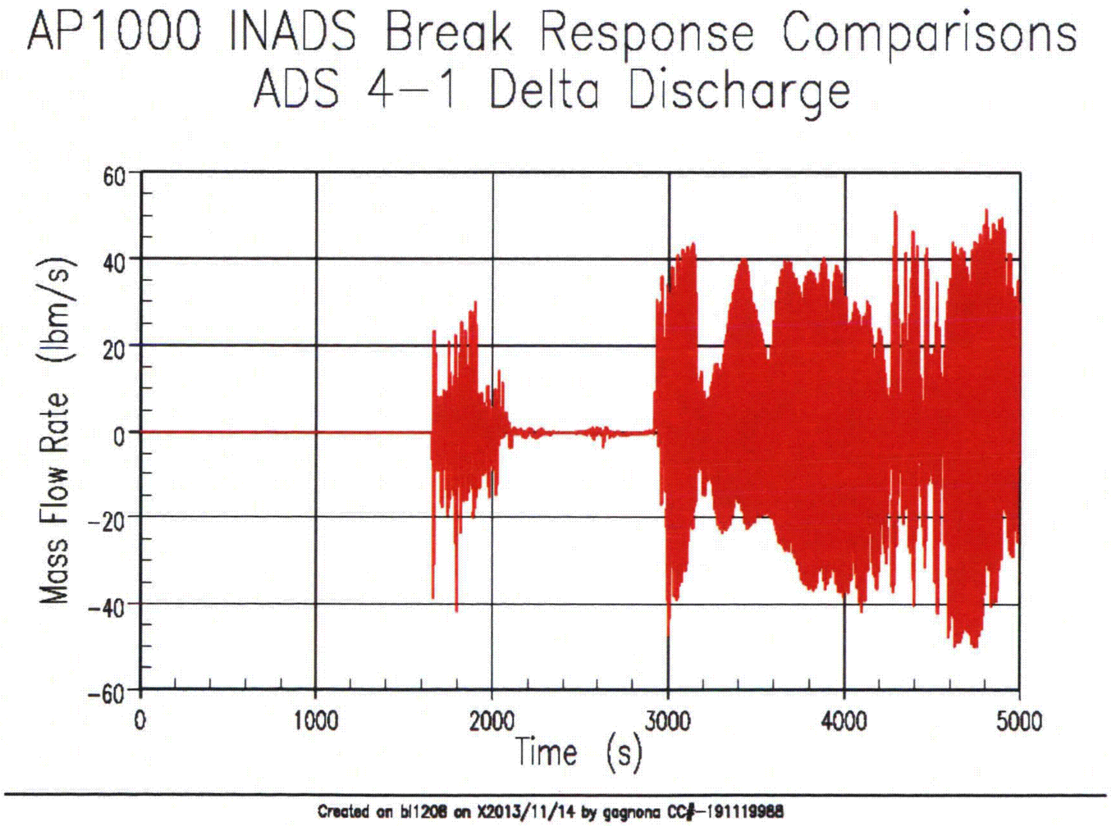
INADS Iteration 1 – ADS 4-1 Discharge Differences



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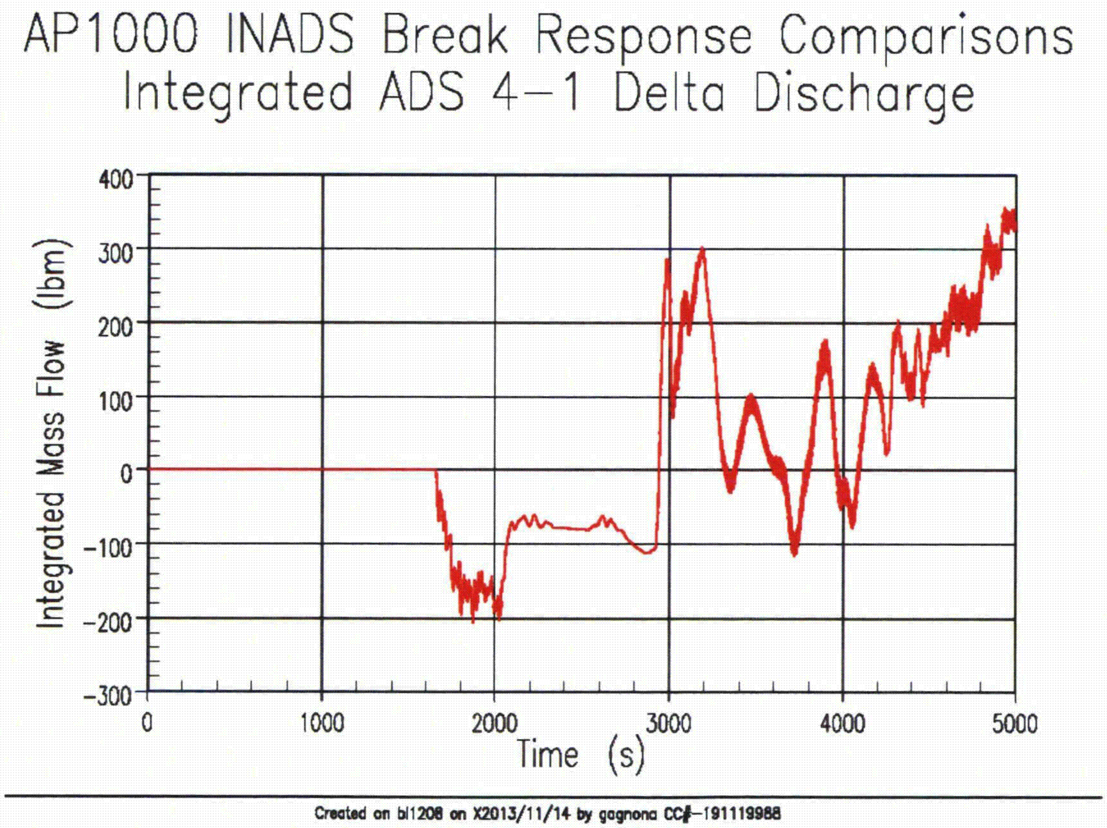
INADS Iteration 1 – ADS 4-1 Discharge Differences



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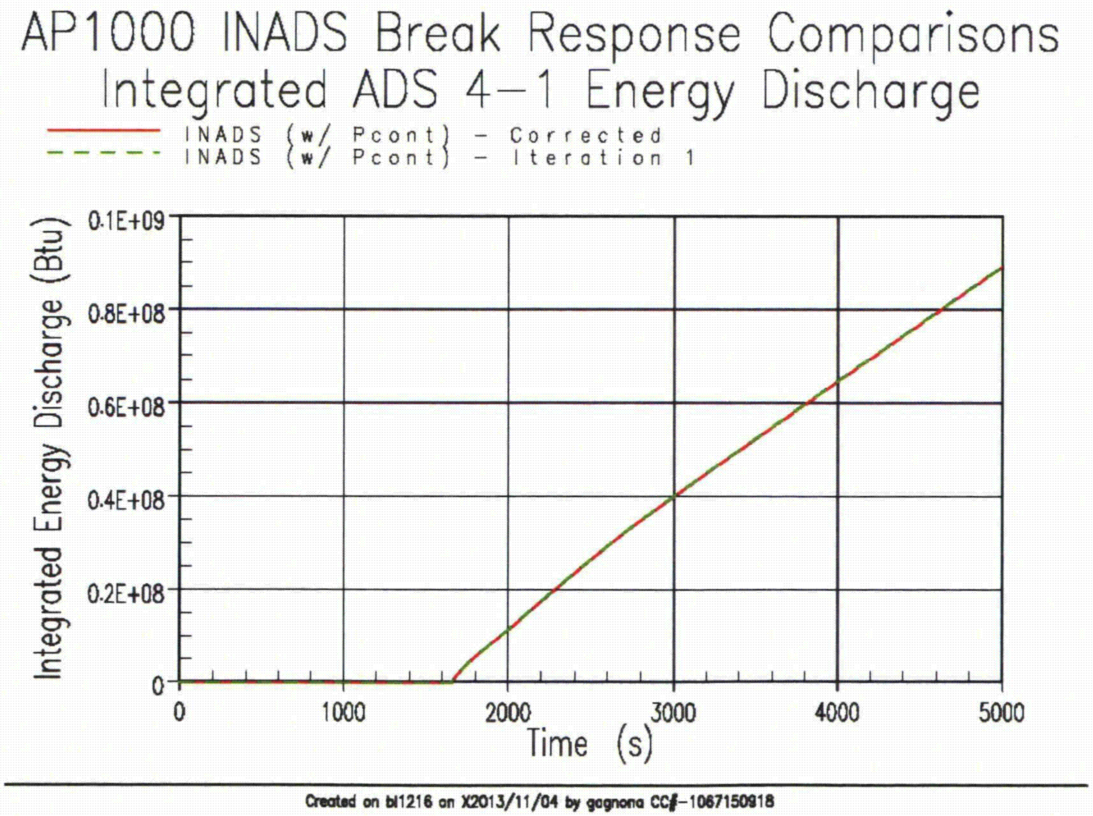
INADS Iteration 1 – Integrated ADS 4-1 Energy Discharge



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Conclusion

- 2 inch results indicate relative insensitivity to iterations
- Additional INADS simulations confirm this observed behavior
- Post IRWST initiation containment response variations have minimal impact on important SBLOCA results
- No additional iterations with containment group considered necessary

Westinghouse

Slides Presented on 11/26/2013

Additional NRC Audit Slides

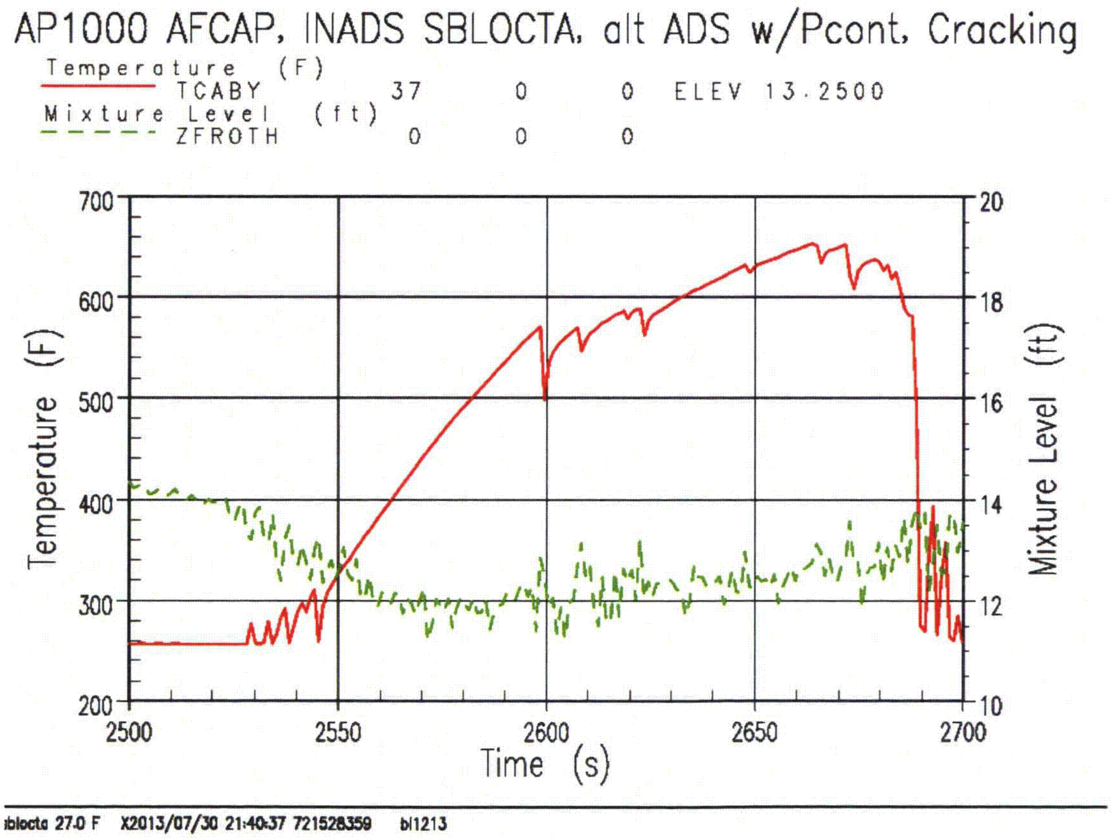
Westinghouse

Additional Plots Requested By NRC

- Inadvertent ADS (INADS) Simulation PCT Plots For Reported Case and Iteration 1
  - PCT and Mixture Level vs. Time For Reported Case
  - PCT and Mixture Level vs. Time For Iteration 1 Case
  - Mixture Level Comparison Plot
- 2 inch Cold Leg Break Containment Pressure Comparison
  - Extracted From Analysis Calc Note
- 2 inch ADS-4 Non-Critical Transition Plots
  - 14.7 psia Containment Pressure
  - Final Variable Containment Pressure

Westinghouse

INADS – Reported PCT Case



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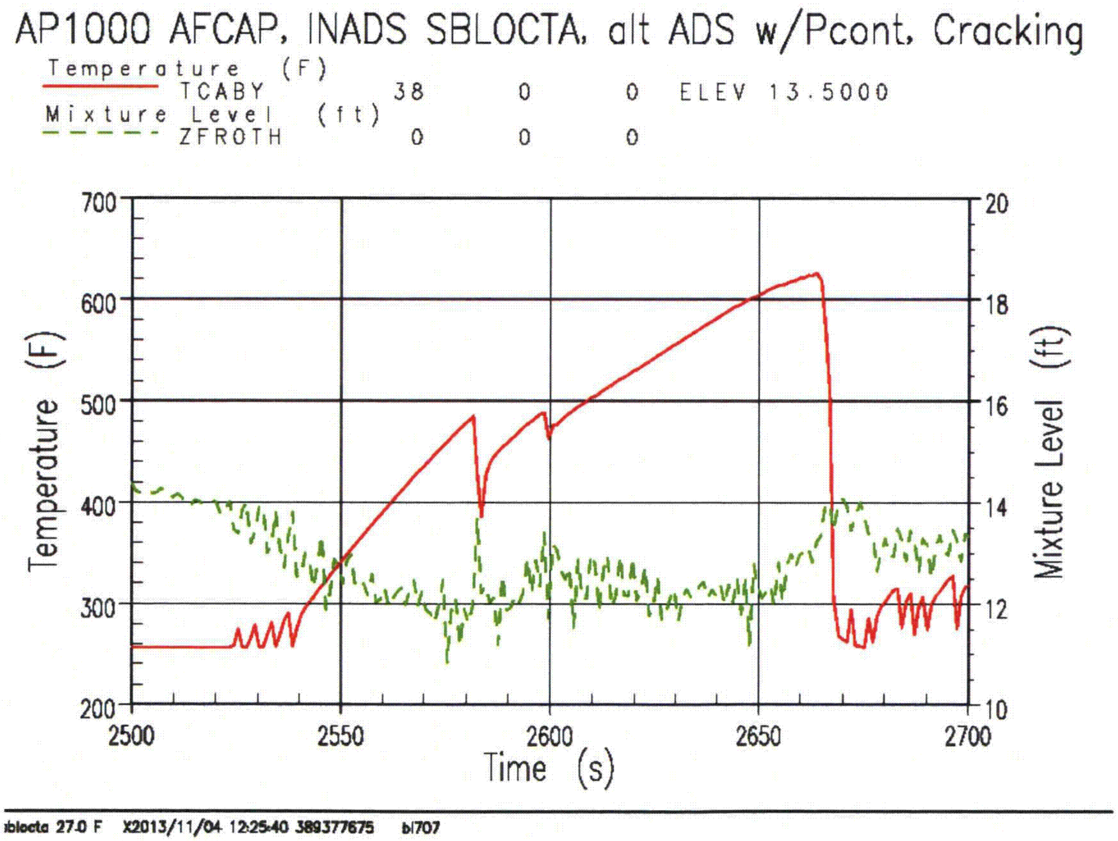
INADS – Iteration 1 PCT Case



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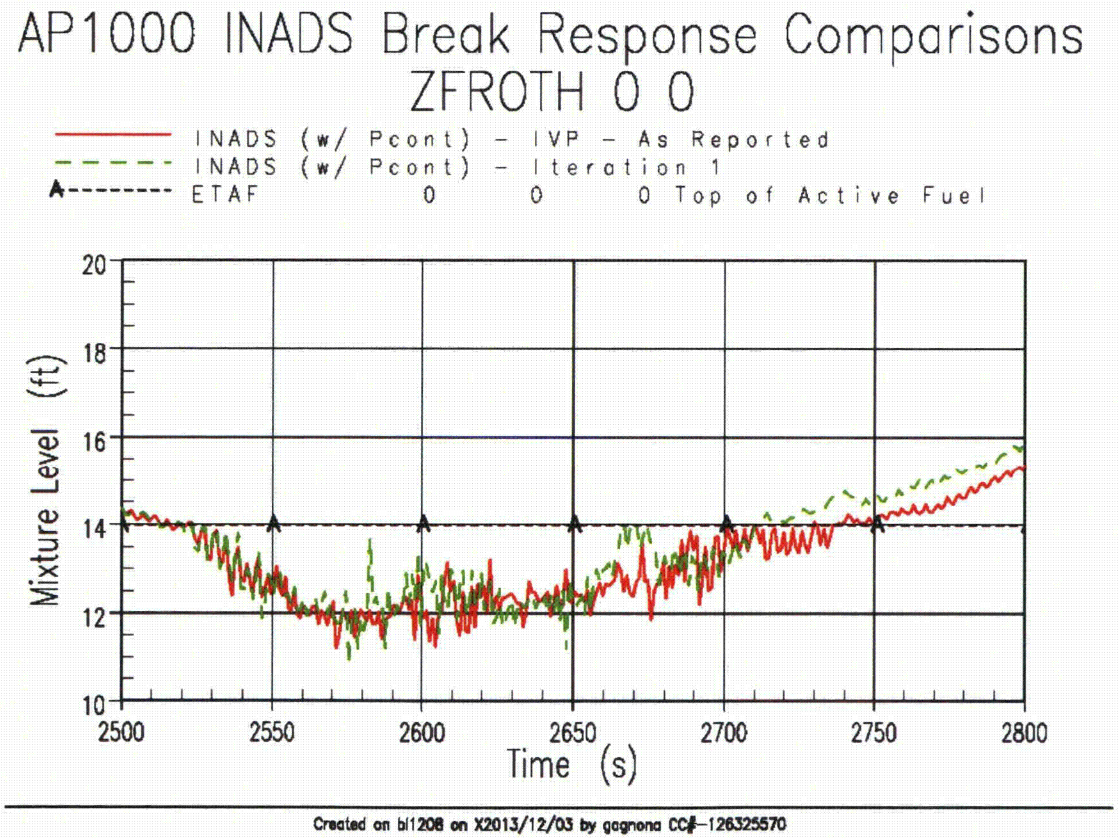
INADS – Compare Reported and Iteration 1 Mixture Levels



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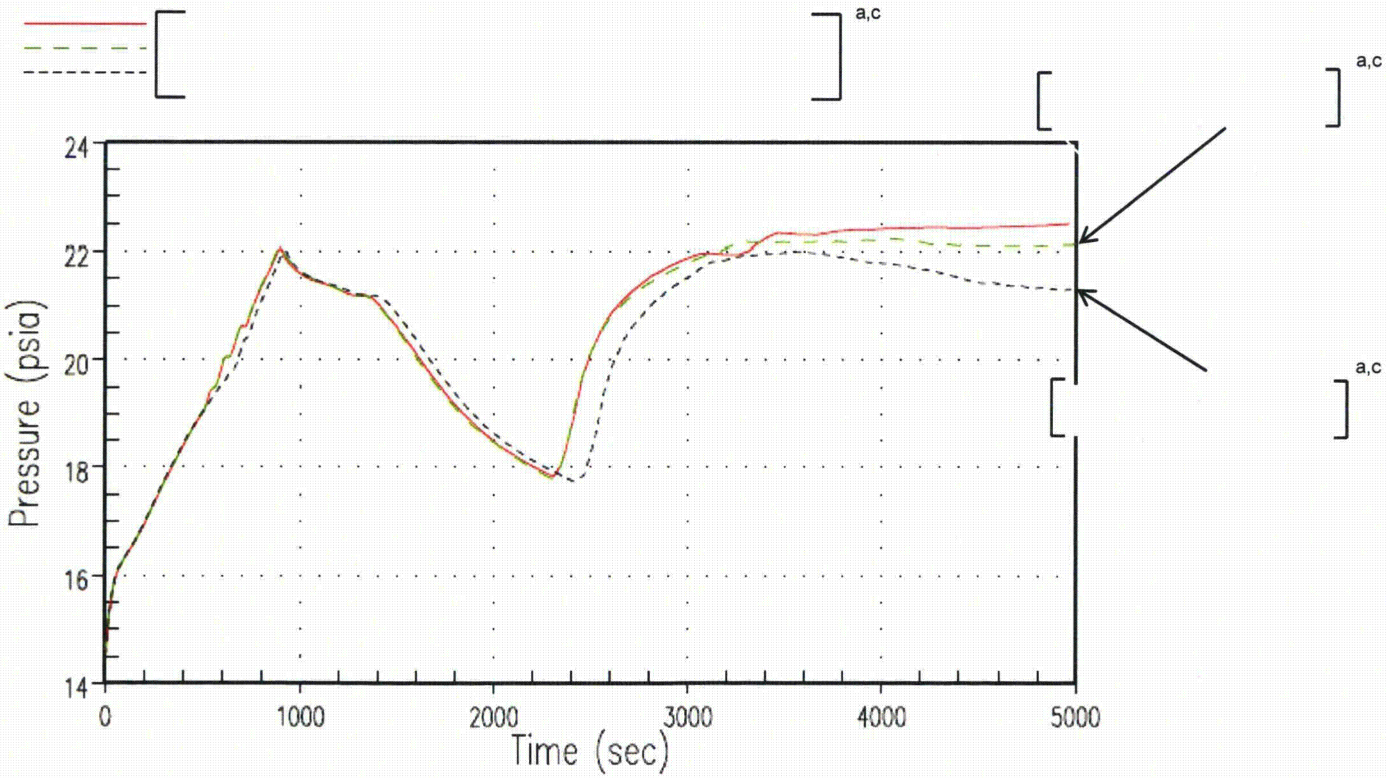
2 inch Cold Leg Break Containment Response



Westinghouse

2 inch Cold Leg Break, 14.7 psia Containment, ADS-4 Non-Critical Transition

a,c

Westinghouse

2 inch Cold Leg Break, Final Variable Containment, ADS-4 Non-Critical Transition

a,c

Westinghouse

SBLOCA CRR Containment Minimum Backpressure Calculation Audit Response for Initial Air Temperature

Michael Patterson

Senior Engineer

Containment and Radiological Analysis

Westinghouse

Overview

- NRC audit question…

  "For Westinghouse to provide a justification for using [ ]a,c for the outside air temperature in the minimum containment pressure calculations."

- The current SBLOCA minimum containment backpressure WGOTHIC analyses assumption of [ ]a,c as the outside initial air temperature is conservative
  - Analysis results are more sensitive to the initial inside air temperature and humidity (higher values are conservative)
    - Reduced outside air temperature requires reduced air temperature inside
  - **No** significant changes in alternate ADS-4 simulation response and M&E's observed with small changes in containment pressure (iteration discussion)

Westinghouse

GOTHIC Modeling

- To illustrate the above conclusion, sensitivities were performed using an AP1000 GOTHIC containment model to confirm use of a lower outside air temperature is non-conservative and dominance of the effect of inside air temperature
- The modeling assumptions are consistent with APP-SSAR-GSC-786 (as well as WCAP-15846 and SER), except:

  [ ] a,c
- Inadvertent ADS and SBLOCA sensitivity cases were run with:

  [ ] a,c

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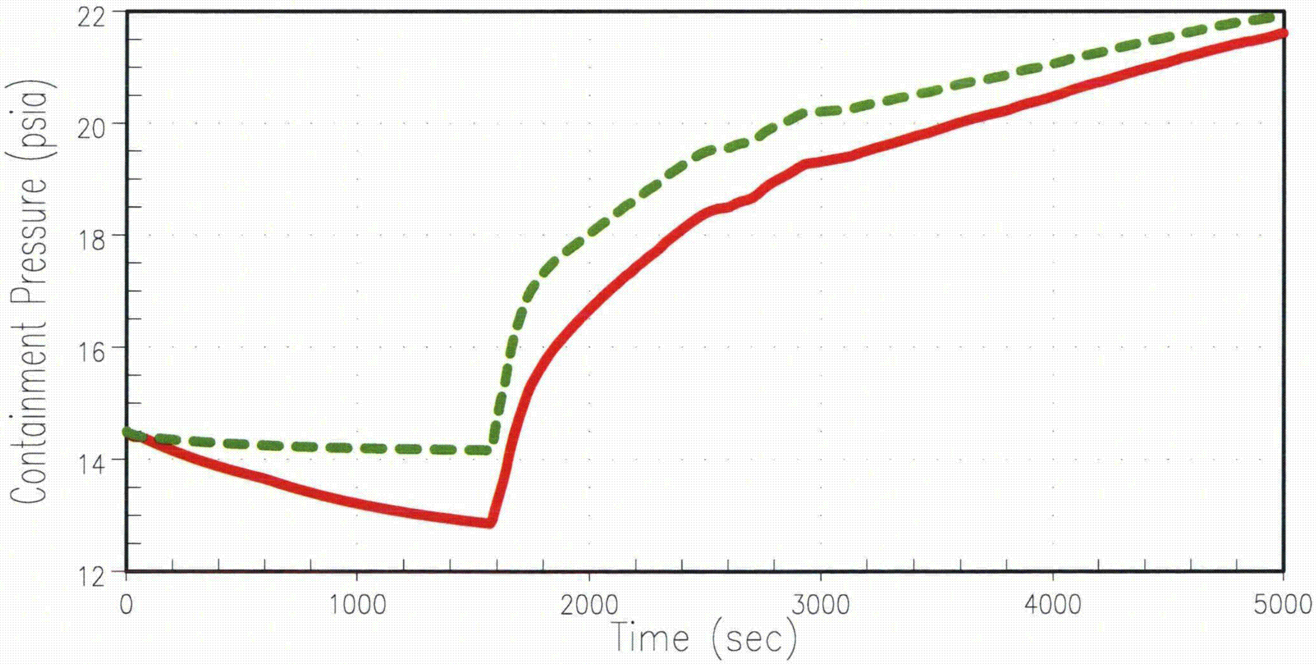
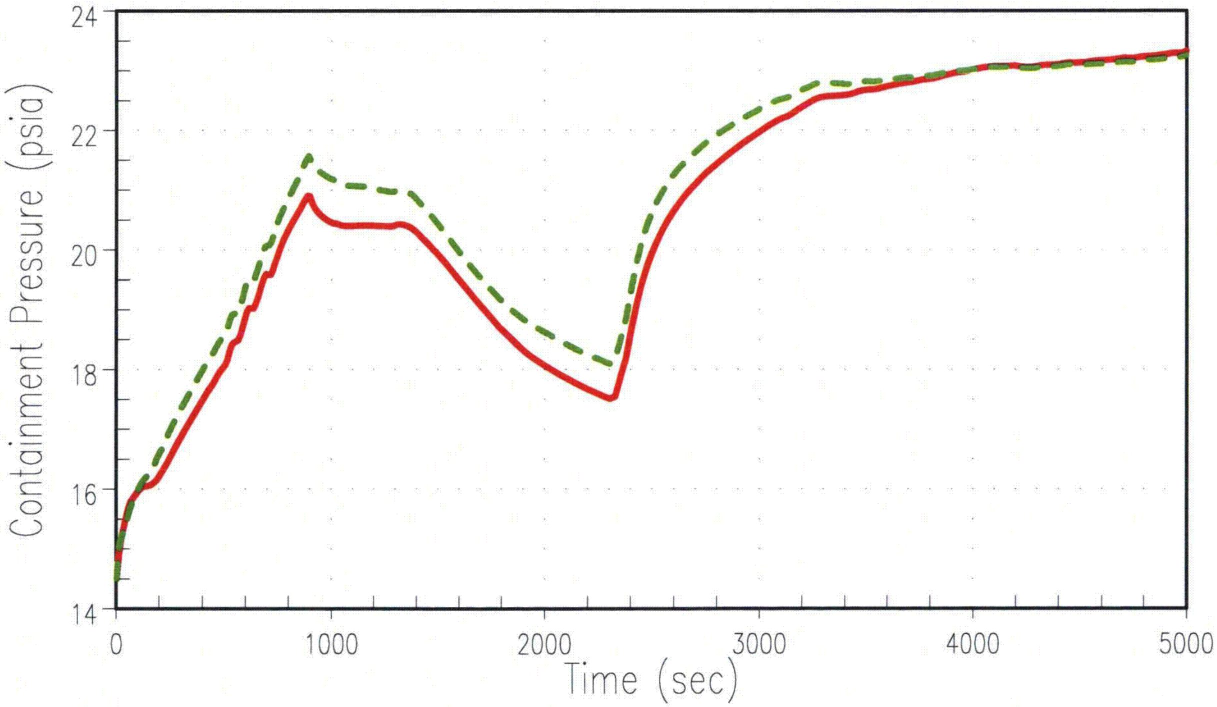
Inadvertent ADS Initial Temp Pressure Comparison

[ ]a,c



2" SBLOCA Initial Temp Pressure Comparison

[ ] a,c



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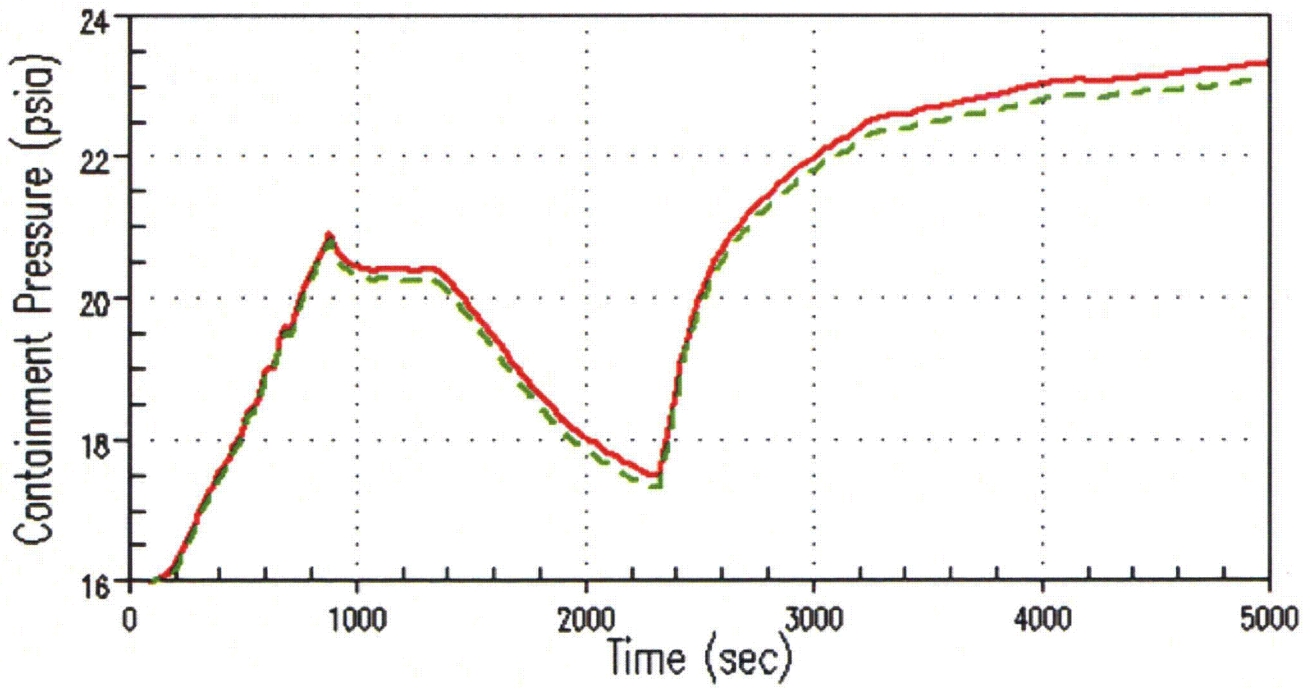
2" SBLOCA Initial Temp Differential Pressure Comparison

[ ]a,c



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Conclusions

- The containment pressure response is sensitive to the initial internal temperature and humidity
  - With higher initial containment temperature and humidity, more steam is able to condense on the containment shell and heat sinks, resulting in a lower transient pressure response
- The use of [ ][a,c] as the initial outside air temperature is conservative compared to [ ][a,c] for the following reasons:
  - At [ ][a,c], the maximum initial containment temperature is limited to [ ][a,c] as opposed to 120°F
  - Once PCS is initiated [ ][a,c], the temperature difference through the containment shell is greater with a higher initial containment temperature
  - The fan coolers would remove more heat at the higher initial containment temperature

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SBLOCA CRR Containment Minimum Backpressure Calculation Audit Response for Impact of IOZ Thermal Conductivity

Michael Patterson

Senior Engineer

Containment and Radiological Analysis

Westinghouse

Overview

- Addendum 1 of WCAP-15846 has no impact on the values or methods used in the AP1000 WGOTHIC minimum containment pressure models.
- WCAP-15846 Addendum 1
  - Develops an effective thermal conductivity value for the IOZ coating in peak containment pressure analyses
  - Confirms the value currently used, does not change the AP1000 WGOTHIC analyses, modeling, or results
- [ ]^{a,c} currently used in our DCD Rev. 19 AP1000 containment peak pressure analyses
  - A lower thermal conductivity is conservative for peak pressure analyses (minimizes heat transfer through the shell)
- A conservatively high value of [ ]^{a,c}, based on approved vendor test value, is used in WGOTHIC minimum containment pressure model
  - [ ]^{a,c}
  - Biased high for maximum heat and mass transfer through the shell
- An additional sensitivity was completed using the approved vendor test value for IOZ increased by 20%
  - Figure on following page shows no impact to the minimum containment backpressure

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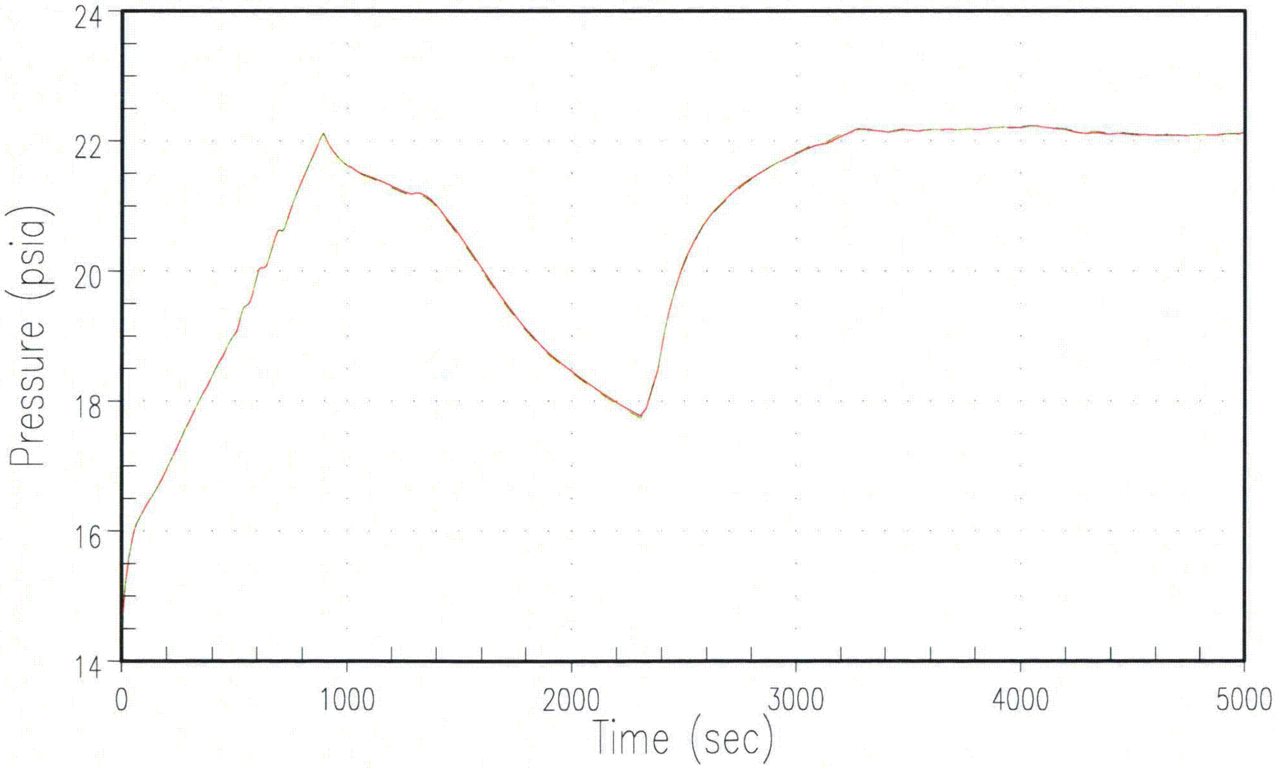
Thermal Conductivity Sensitivity

[ ]a,c



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Section K

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LTR-NRC-14-20

April 4, 2014

Subject: Supplemental Information on the Impacts of Errors in the Loss of Coolant Accident Evaluation Models (Non-Proprietary).

As part of WCAP-17524 Revision 0, "AP1000 Core Reference Report" the analyses for Large Break Loss of Coolant Accident (LBLOCA) and Small Break Loss of Coolant Accident (SBLOCA) were provided to the NRC for review and approval. Subsequent to the submittal of WCAP-17524 Revision 0, the LBLOCA and SBLOCA analyses were revised and the revised analyses are presented in WCAP-17524 Revision 1. The purpose of this letter is to provide supplemental information for the **AP1000**® plant LBLOCA and SBLOCA analyses presented in Revision 1 of the Core Reference Report (CRR).

If you have any questions or require additional information, please contact Keith Drudy at (412) 374-5841.

Very truly yours,

William J. Snoody / FOR

James A. Gresham, Manager

Regulatory Compliance

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Supplemental Information on the Impacts of Errors in the Loss of Coolant Accident Evaluation Models (Non-Proprietary)

April 2014

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Cranberry Township, PA 16066

Supplemental Information on the Impacts of Errors in the Loss of Coolant Accident Evaluation Models

As part of WCAP-17524 Revision 0, "AP1000 Core Reference Report" the analyses for Large Break Loss of Coolant Accident (LBLOCA) and Small Break Loss of Coolant Accident (SBLOCA) were provided to the NRC for review and approval. Subsequent to the submittal of WCAP-17524 Revision 0, the LBLOCA and SBLOCA analyses were revised and the revised analyses are presented in WCAP-17524 Revision 1.

The purpose of this letter is to provide supplemental information for the **AP1000®** plant LBLOCA and SBLOCA analyses presented in Revision 1 of the Core Reference Report (CRR). Errors have been identified in the evaluation models for the LBLOCA and SBLOCA analyses presented in Revision 1 of the CRR. To address these errors, as well as changes to the evaluation models, the process described in 10 CFR 50.46 has been utilized. The supplemental information provided in Attachment 1 consists of 10 CFR 50.46 reporting sheets and current peak cladding temperature (PCT) rackup sheets for the LBLOCA and SBLOCA analyses presented in Revision 1 of the CRR. The 10 CFR 50.46 reporting sheets describe known changes to or errors in the LOCA evaluation models applied in the CRR analyses, and the estimated PCT effect of each change or error.

For the LBLOCA ASTRUM evaluation model, the included 10 CFR 50.46 reporting sheets consist of changes and errors reported after the 2007 reporting year which are applicable to the LBLOCA CRR Revision 1 analysis. It is noted that applicable 10 CFR 50.46 reporting pages for the 1998 through 2007 reporting years were previously provided to the NRC via APP-GW-GLE-026, Revision 1 as part of the licensing of the ASTRUM evaluation model application to the **AP1000** plant.

For the LBLOCA evaluation model there are several 0 degree errors and 2 non-zero errors, with a cumulative estimated PCT impact of 34 °F relative to the PCT documented in CRR Revision 1.

For the SBLOCA NOTRUMP-AP600 evaluation model, the included 10 CFR 50.46 reporting sheets consist of changes and errors reported after the **AP1000** plant final safety evaluation report (NUREG-1793, Initial Report) was issued, for reporting years from 2004 through 2013, which are applicable to the SBLOCA CRR Revision 1 analysis.

For the SBLOCA evaluation model there are several 0 degree errors and 0 non-zero errors, with a cumulative estimated PCT impact of 0 °F relative to the analysis documented in CRR Revision 1.

Attachment 1

2008 REPORTING

NOTRUMP BUBBLE RISE/DRIFT FLUX MODEL INCONSISTENCIES

Background
The standard plant version of the NOTRUMP code was updated to resolve inconsistencies in several drift flux models as well as the nodal bubble rise/droplet fall models. In summary, these changes include:

- Bubble rise and droplet fall model calculations were made consistent with flow link calculations.
- Corrections were made to limits employed in the vertical counter-current flooding models.
- Checking logic was added to correct situations where drift flux model inconsistencies could result (i.e. prevent liquid flow from an all-vapor node and vapor flow from all-liquid node).

All of these changes represent a closely-related group of Non-Discretionary Changes in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Models
1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
Due to the nature of the Advanced Plant SBLOCA response and the relative insensitivity of these corrections on the Standard Plant EM, it is estimated that the effect of these inconsistencies on PCT calculations is 0°F for 10 CFR 50.46 purposes for the Advanced Plant NOTRUMP EM calculations.

2008 REPORTING

NOTRUMP DRIFT FLUX MODEL INCONSISTENCIES

Background
The standard plant version of the NOTRUMP code was updated to resolve inconsistencies in the resetting of certain parameters in the drift flux models when single phase conditions are determined to exist. The previous coding had inadvertently omitted certain conditions on drift velocity and void fraction which are now included. Also, in the node boundary mixture level crossing logic, several partial derivatives for liquid and vapor volumetric fluxes with respect to mass flux in the void fraction model were erroneously set to zero. The correct partial derivative calculations were added to the code. In addition, several instances (stacking logic, accumulator empty logic and pump critical flow logic) where flow link specific volumes were incorrectly always based on saturated conditions were corrected. These changes represent a closely-related group of Non-Discretionary Changes in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Models
1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
The subject changes involve logic that is seldom used in standard plant EM calculations. As such, the estimated effect of these inconsistencies on PCT calculations is 0°F for 10 CFR 50.46 reporting purposes for the Advanced Plant NOTRUMP EM calculations.

2008 REPORTING

NOTRUMP INVERTED T-NODE SIGN CONVENTION

Background
This change deals with the correction of the sign convention for inverted T-nodes, which was incorrectly applied via input into the EM. It can potentially impact the reactor vessel lower plenum node, Passive Residual Heat Removal (PRHR) inlet piping and ADS 1-3 discharge piping in the Advanced Plant EM. This change represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Models
1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
The correction of this error can affect the mixture/vapor interfacial area within a fluid node. With respect to the potentially impacted areas of the model, this condition never exists in the reactor vessel lower plenum. It is judged that the impact of this error correction on the other areas is insignificant on the overall results. Based on this judgment, coupled with the fact that plant model calculations show this to be the case, the correction of this error will be assigned a 0°F PCT impact for 10 CFR 50.46 reporting purposes.

2008 REPORTING

NOTRUMP VAPOR REGION FORMATION LOGIC

Background
The logic governing formation of a vapor region within a fluid node in NOTRUMP was corrected in the standard plant code to allow superheated conditions where appropriate, instead of saturated conditions which may not exist at that instant. This change represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Models
1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
Typically, region formation conditions in standard plant EM calculations occur at saturation in all control volumes outside of the core model. If a region is formed at superheat conditions in these areas of the model, the amount of superheat is usually small. As such, the nature of these logic issues leads to an estimated PCT impact of 0°F for 10 CFR 50.46 reporting purposes for the Advanced Plant NOTRUMP EM calculations.

2008 REPORTING

CHECK VALVE IN SERIES WITH AN ISOLATION VALVE

Background

During review of preliminary analyses performed for the **AP1000®** plant analysis program, reverse flows were observed through the Core Makeup Tank (CMT) check valves under inappropriate conditions. The conditions where this behavior was observed occurred when a negative pressure differential across the check valves existed concurrent with the opening of the isolation valves. A review of the flow links in the NOTRUMP AP600 and **AP1000** plant models indicate that the following flow paths could be affected:

- Core Makeup Tank (CMT) Discharge lines. These paths are modeled with isolation valves in series with check valves w/drift flux.
- In-Containment Refueling Water Storage Tank (IRWST) Injection lines. These paths are modeled with isolation valves in series with check valves and no drift flux.
- Passive Residual Heat Removal (PRHR) discharge line. This path is modeled with an isolation valve w/drift flux.

A cursory review of both the AP600 and **AP1000** plant analyses indicates that only transients with a negative pressure differential across the check valves while the isolation valves are opening (i.e. 10-inch and Cold Leg Balance Line (CLBL) breaks) exhibit the potential to predict a period of negative flow. Due to the coding logic utilized, the concern only exists while the isolation valves are opening which occurs for a short duration and will not likely have a significant impact on results of the SBLOCA analyses.

When drift flux is applied to a variable area flow path (i.e., a path containing a check valve, isolation valve, or a combination) the code may predict flow behavior not appropriate for the actual conditions. A temporary workaround of modeling the flow paths as homogenous is acceptable due to the subcooled nature of the affected flow paths (CMT discharge and PRHR heat exchanger discharge).

These changes represent a closely related group of Non-Discretionary Changes in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)

1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect

The potential for negative flow through a check valve in series with an isolation valve only exists during the short period over which the isolation valve is opening and as such the impact on transient results is minimal. The subcooled nature of the flow through the variable area flow paths with drift flux applied is expected to lead to inappropriate flow having a minimal change in the overall transient results. Thus, for Advanced Plant small break LOCA analysis results, the estimated PCT impact is 0°F for 10 CFR 50.46 reporting purposes.

2008 REPORTING

GENERAL CODE MAINTENANCE

Background
Various changes have been made to enhance the usability of the codes and to help preclude errors in analyses. This includes items such as modifying input variable definitions, units, and defaults; improving the input diagnostic checks; enhancing the code output; optimizing active coding; and, eliminating inactive coding. These changes represent Discretionary Changes that will be implemented on a forward-fit basis in accordance with Section 4.1.1 of WCAP-13451.

Affected Evaluation Model(s)
1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
The nature of these changes leads to an estimated PCT impact of 0°F.

2009 REPORTING

ERROR IN ASTRUM PROCESSING OF AVERAGE ROD BURNUP AND ROD INTERNAL PRESSURE

Background
An error was discovered in the processing of the burnup and rod internal pressure inputs for average core rods in ASTRUM analyses. The correction of this error has been evaluated for impact on current licensing-basis analyses and will be incorporated into the ASTRUM method at a future time. These changes represent a closely-related group of Non-Discretionary Changes in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect
This error was evaluated to have a negligible impact on PCT, leading to an estimated impact of 0°F for 10 CFR 50.46 reporting purposes.

2011 REPORTING

GENERAL CODE MAINTENANCE

Background
Various changes have been made to enhance the usability of codes and to streamline future analyses. Examples of these changes include modifying input variable definitions, units and defaults; improving the input diagnostic checks; enhancing the code output; optimizing active coding; and eliminating inactive coding. These changes represent Discretionary Changes that will be implemented on a forward-fit basis in accordance with Section 4.1.1 of WCAP-13451.

Affected Evaluation Model(s)
1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model
1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model, Application to PWRs with Upper Plenum Injection
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect
The nature of these changes leads to an estimated PCT impact of 0°F.

2011 REPORTING

ERROR IN VARIOUS REGION ELEVATIONS AND SUBSEQUENT RELATED CALCULATIONS

Background
Several closely related errors associated with component elevations were discovered in the **AP1000®** plant data collection. Some of these component elevations are subsequently utilized in the calculation of region volumes, surface areas and metal masses which are used in safety analyses.

This represents a Non-Discretionary Change to the Evaluation Model as described at Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)
1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
This issue was judged to have a negligible impact on existing analyses, leading to an estimated Peak Cladding Temperature (PCT) impact of 0°F.

2012 REPORTING

GENERAL CODE MAINTENANCE

Background
Various changes have been made to enhance the usability of codes and to streamline future analyses. Examples of these changes include modifying input variable definitions, units and defaults; improving the input diagnostic checks; enhancing the code output; optimizing active coding; and eliminating inactive coding. These changes represent Discretionary Changes that will be implemented on a forward-fit basis in accordance with Section 4.1.1 of WCAP-13451.

Affected Evaluation Model(s)
1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model
1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model for Application to PWRs with Upper Plenum Injection
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect
The nature of these changes leads to an estimated Peak Cladding Temperature (PCT) impact of 0°F.

2012 REPORTING

HOTSPOT BURST TEMPERATURE CALCULATION FOR ZIRLO CLADDING

Background
A problem was identified in the calculation of the burst temperature for **ZIRLO®** cladding in the HOTSPOT code when the cladding engineering hoop stress exceeds 15,622 psi. This problem results in either program failure or an invalid extrapolation of the burst temperature vs. engineering hoop stress table. This problem has been evaluated for impact on existing analyses, and its resolution represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)
1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model
1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model for Application to PWRs with Upper Plenum Injection
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect
The evaluation of existing analyses demonstrated no impact on the overall Peak Cladding Temperature (PCT) results, leading to an estimated effect of 0°F.

2012 REPORTING

HOTSPOT ITERATION ALGORITHM FOR CALCULATING THE INITIAL FUEL PELLET AVERAGE TEMPERATURE

Background
The HOTSPOT code has been updated to incorporate the following corrections to the iteration algorithm for calculating the initial fuel pellet average temperature: (1) bypass the iteration when the input value satisfies the acceptance criterion; (2) prevent low-end extrapolation of the gap heat transfer coefficient; (3) prevent premature termination of the iteration that occurred under certain conditions; and (4) prevent further adjustment of the gap heat transfer coefficient after reaching the iteration limit. These changes represent a closely-related group of Non-Discretionary Changes in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)
1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model
1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model for Application to PWRs with Upper Plenum Injection
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect
Sample calculations and engineering judgment lead to an estimated Peak Cladding Temperature (PCT) impact of 0°F.

2012 REPORTING

WCOBRA/TRAC AUTOMATED RESTART PROCESS LOGIC ERROR

Background
A minor error was identified in the WCOBRA/TRAC Automated Restart Process (WARP) logic for defining the Double-Ended Guillotine (DEG) break tables. The error has been evaluated for impact on current licensing-basis analysis results and will be incorporated into the plant-specific analyses on a forward-fit basis. These changes represent a closely-related group of Non-Discretionary Changes in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)
1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model
1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model for Application to PWRs with Upper Plenum Injection
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect
These errors were evaluated to have a negligible impact on the Large Break LOCA analysis results, leading to an estimated Peak Cladding Temperature (PCT) impact of 0°F.

2012 REPORTING

ROD INTERNAL PRESSURE CALCULATION

Background

Several issues which affect the calculation of rod internal pressure (RIP) have been identified for certain Best-Estimate (BE) Large-Break Loss-of-Coolant Accident (LBLOCA) evaluation models (EMs). These issues include the sampling of rod internal pressure uncertainties, updating HOTSPOT to consider the effect of transient RIP variations in the application of the uncertainty, and generating RIPs at a consistent rod power. These issues have been evaluated to estimate the impact on existing LBLOCA analysis results. The resolution of these issues represents a closely-related group of Non-Discretionary Changes in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

The effects described above are either judged to have a negligible effect on existing LBLOCA analysis results or have been adequately incorporated into the thermal conductivity degradation evaluations, leading to an estimated Peak Cladding Temperature (PCT) impact of 0°F.

2012 REPORTING

WCOBRA/TRAC THERMAL-HYDRAULIC HISTORY FILE DIMENSION USED IN HSDRIVER

Background

A problem was identified in the dimension of the WCOBRA/TRAC thermal-hydraulic history file used in HSDRIVER. The array that is used to store the information from the WCOBRA/TRAC thermal-hydraulic history file is dimensioned to 3000 in HSDRIVER. It is possible for this file to contain more than 3000 curves. If that is the case, it is possible that the curves would not be used correctly in the downstream HOTSPOT execution. An extent-of-condition review indicated that resolution of this issue does not impact the Peak Cladding Temperature (PCT) calculation for prior Large Break Loss-of-Coolant Accident (LBLOCA) analyses. This represents a Discretionary Change in accordance with Section 4.1.1 of WCAP -13451.

Affected Evaluation Model(s)

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model
1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model for Application to PWRs with Upper Plenum Injection
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

As discussed in the Background section, resolution of this issue does not impact the PCT calculation for prior LBLOCA analyses, which leads to a PCT impact of 0°F.

2012 REPORTING

ERRORS IN VARIOUS COMPONENT ELEVATIONS AND METAL MASSES

Background
Several closely related errors associated with various component elevations and metal masses were discovered in the **AP1000**® plant calculations. These errors have been evaluated and will be corrected in the future. These changes represent a closely-related group of Non-Discretionary Changes in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)
1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
These errors were evaluated as having a negligible impact on existing analyses, leading to an estimated Peak Cladding Temperature (PCT) impact of 0°F.

2013 REPORTING

GENERAL CODE MAINTENANCE

Background
Various changes have been made to enhance the usability of codes and to streamline future analyses. Examples of these changes include modifying input variable definitions, units and defaults; improving the input diagnostic checks; enhancing the code output; optimizing active coding; and eliminating inactive coding. These changes represent Discretionary Changes that will be implemented on a forward-fit basis in accordance with Section 4.1.1 of WCAP-13451.

Affected Evaluation Model(s)
1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect
The nature of these changes leads to an estimated Peak Cladding Temperature (PCT) impact of 0°F.

2013 REPORTING

BURST ELEVATION SELECTION

Background
It is stated on page 11-20 of WCAP-16009-P-A that the burst option is applied at the elevation corresponding to the (WCOBRA/TRAC) burst elevation for the hot assembly rod. This approach was modified to apply the burst option at the HOTSPOT predicted burst elevation as described on page 19 of Attachment 1 to LTR-NRC-06-8. The HOTSPOT code has been updated to incorporate the following changes to the burst elevation selection logic if multiple nodes burst at the same time: (1) the node that has the highest cladding temperature at the time of burst is selected; (2) if multiple nodes have the same burst time and cladding temperature at the time of burst, the lowest ordered elevation of those nodes is selected. These changes represent a closely-related group of Discretionary Changes in accordance with Section 4.1.1 of WCAP-13451.

Affected Evaluation Model(s)
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect
This improvement in burst elevation selection is a forward-fit change, leading to an estimated Peak Cladding Temperature (PCT) impact of 0°F.

2013 REPORTING

ELEVATIONS FOR HEAT SLAB TEMPERATURE INITIALIZATION

Background
An error was discovered in WCOBRA/TRAC whereby an incorrect value would be used in the initial fuel rod temperature calculation for a fuel rod heat transfer node if that node elevation was specified outside of the bounds of the temperature initialization table. This problem has been evaluated for impact on existing analyses and its resolution represents a Discretionary Change in accordance with Section 4.1.1 of WCAP-13451.

Affected Evaluation Model(s)
1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect
Based on inspection of plant analysis input, it was concluded that the input decks for existing analyses are not impacted by this error, leading to an estimated peak cladding temperature impact of 0°F.

2013 REPORTING

HEAT TRANSFER LOGIC CORRECTION FOR ROD BURST CALCULATION

Background
A change was made to the WCOBRA/TRAC coding to correct an error which had disabled rod burst in separate effect test simulations. This change represents a Discretionary Change in accordance with Section 4.1.1 of WCAP-13451.

Affected Evaluation Model(s)
1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect
Based on the nature of the change and the evaluation model requirements for plant modeling in Westinghouse best estimate large break LOCA analyses with WCOBRA/TRAC, it is judged that existing analyses are not impacted by this change, leading to an estimated peak cladding temperature impact of 0°F.

2013 REPORTING

WCOBRA/TRAC U19 FILE DIMENSION ERROR CORRECTION

Background

A problem was identified in the dimension of an array used to generate the u19 file in WCOBRA/TRAC. The u19 file is read during HSDRIVER execution and provides information needed to generate the HOTSPOT thermal-hydraulic history and user input files. The array used to write the desired information to the u19 file is dimensioned to 2000 in WCOBRA/TRAC. It is possible, however, for more than 2000 curves to be written to the u19 file. If that is the case, it is possible that the curves would not be stored correctly on the u19 file. A survey of current Best Estimate Large Break LOCA analyses indicated that the majority of plants had less than 2000 curves in their u19 files; therefore these plants are not affected by the change. For those plants with more than 2000 curves, plant-specific sensitivity calculations indicated that resolution of this issue does not impact the peak cladding temperature (PCT) calculation for prior analyses. This represents a Discretionary Change in accordance with Section 4.1.1 of WCAP-13451.

Affected Evaluation Model(s)

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

As discussed in the Background section, resolution of this issue does not impact the peak cladding temperature calculation for prior LBLOCA analyses, leading to an estimated peak cladding temperature impact of 0°F.

2013 REPORTING

HEAT TRANSFER MODEL ERROR CORRECTIONS

Background

Several related changes were made to WCOBRA/TRAC to correct errors discovered which affected the heat transfer models. These errors included calculation of the entrained liquid fraction used in calculation of the drop wall heat flux, application of the grid enhancement factor for grid temperature calculation, calculation of the Reynold's number used in the Wong-Hochrieter correlation for the heat transfer coefficient from fuel rods to vapor, fuel rod initialization and calculation of cladding inner radius with creep, application of grid and two phase enhancement factors and radiation component in single phase vapor heat transfer, and reset of the critical heat flux temperature when J=2. These errors have been evaluated to estimate the impact on existing LBLOCA analysis results. Correction of these errors represents a closely-related group of Non-Discretionary Changes in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

Based on the results of representative plant calculations, separate effects and integral effects test simulations, it is concluded that the error corrections have a negligible local effect on heat transfer, leading to an estimated peak cladding temperature impact of 0°F.

2013 REPORTING

CORRECTION TO HEAT TRANSFER NODE INITIALIZATION

Background

An error was discovered in the heat transfer node initialization logic in WCOBRA/TRAC whereby the heat transfer node center locations could be inconsistent with the geometric node center elevations. The primary effects of this issue are on the interpolated fluid properties and grid turbulent mixing enhancement at the heat transfer node. This problem has been evaluated for impact on existing analyses and its resolution represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

Based on engineering judgment and the results from a matrix of representative plant calculations, it is concluded that the effect of this error is within the code resolution, leading to an estimated peak cladding temperature impact of 0°F.

2013 REPORTING

MASS CONSERVATION ERROR FIX

Background
It was identified that mass was not conserved in WCOBRA/TRAC one-dimensional component cells when void fraction values were calculated to be slightly out of the physical range (greater than 1.0 or smaller than 0.0). This was observed to result in artificial mass generation on the secondary side of steam generator components. Correction of this problem represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)
1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect
This error was observed to primarily affect the mass on the secondary side of the steam generator. This issue was judged to have a negligible impact on existing LBLOCA analysis results, leading to an estimated peak cladding temperature impact of 0°F.

2013 REPORTING

CORRECTION TO SPLIT CHANNEL MOMENTUM EQUATION

Background

An error was discovered in the momentum equation calculations for split channels in WCOBRA/TRAC. This error impacts the (1) continuity area of the phantom/boundary bottom cell; (2) bottom and top continuity area correction factors for the channel inlet at the bottom of a section and for the channel outlet at the top of a section; and (3) drop entrainment mass rate per unit volume and drop de-entrainment mass rate per unit volume contributions to the momentum calculations for split channels. This problem has been evaluated for impact on existing analyses and its resolution represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

Based on the results from a matrix of representative plant calculations, it is concluded that the effect of this error on the quantities directly impacted by the momentum equation calculations for split channels (velocities, flows, etc.) is negligible, leading to an estimated peak cladding temperature impact of 0°F.

2013 REPORTING

CHANGES TO VESSEL SUPERHEATED STEAM PROPERTIES

Background

Several related changes were made to the WCOBRA/TRAC coding for the vessel super-heated water properties, including updating the HGAS subroutine coding to be consistent with WCAP-12945-P-A Equation 10-6, updating the approximation of the enthalpy in the TGAS subroutine to be consistent with the HGAS subroutine coding, and updating the temperature iteration method and convergence criteria in the TGAS subroutine. These changes represent a closely-related group of Non-Discretionary Changes in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

The updates to the calculations of the superheated steam properties had generally less than 1°F impact on the resulting steam temperature values, leading to an estimated peak cladding temperature impact of 0°F.

2013 REPORTING

UPDATE TO METAL DENSITY REFERENCE TEMPERATURES

Background
It was identified that for one-dimensional components in which heat transfer to stainless steel 304 or 316 is modeled, the reference temperature for the metal density calculation was allowed to vary; as a result the total metal mass was not preserved. Correction of this problem represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)
1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect
This change primarily impacts the reactor coolant system loop piping modeled in the large break loss-of-coolant accident (LBLOCA) WCOBRA/TRAC models. It was judged that the effect of this change on the peak cladding temperature results was negligible, leading to an estimated peak cladding temperature impact of 0°F.

2013 REPORTING

DECAY HEAT MODEL ERROR CORRECTIONS

Background

The decay heat model in the WCOBRA/TRAC code was updated to correct the erroneously coded value of the yield fraction directly from fission for Group 19 of Pu-239, and to include the term for uncertainty in the prompt energy per fission in the calculation of the decay heat power uncertainty. Correction of these errors represents a closely-related group of Non-Discretionary Changes in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

These changes have a negligible impact on the calculated decay heat power, leading to an estimated peak cladding temperature impact of 0°F.

2013 REPORTING

CORRECTION TO THE PIPE EXIT PRESSURE DROP ERROR

Background
An error was discovered in WCOBRA/TRAC whereby the frictional pressure drop at the split break TEE connection to the BREAK component was incorrectly calculated using the TEE hydraulic diameter instead of the BREAK component length input. This error has been evaluated for impact on existing analyses and its resolution represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)
1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect
Based on the results from a matrix of representative plant calculations, it is concluded that the effect of this error on the pressure at the break and the break flow is negligible, leading to an estimated peak cladding temperature impact of 0°F.

2013 REPORTING

GRID HEAT TRANSFER ENHANCEMENT CALCULATION

Background

An issue was identified which could affect the calculation of the heat transfer at gridded elevations for Best-Estimate (BE) Large-Break Loss-of-Coolant Accident (LBLOCA) Evaluation Models (EMs). For a specific input condition, the grid heat transfer enhancement factor is calculated based on an erroneous core geometry, which can cause an over-prediction of the heat transfer coefficient at gridded elevations. This issue has been evaluated to estimate the impact on existing LBLOCA analysis results. The resolution of this issue represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

The effect described above was judged to have a negligible effect on existing LBLOCA analysis results, leading to an estimated Peak Cladding Temperature (PCT) impact of 0°F.

2013 REPORTING

REVISED HEAT TRANSFER MULTIPLIER DISTRIBUTIONS

Background

Several changes and error corrections were made to WCOBRA/TRAC and the impacts of these changes on the heat transfer multiplier uncertainty distributions were investigated. During this investigation, errors were discovered in the development of the original multiplier distributions, including errors in the grid locations specified in the WCOBRA/TRAC models for the G2 Refill and G2 Reflood tests, and errors in processing test data used to develop the reflood heat transfer multiplier distribution. Therefore, the blowdown heatup, blowdown cooling, refill, and reflood heat transfer multiplier distributions were redeveloped. For the reflood heat transfer multiplier development, the evaluation time windows for each set of test experimental data and each test simulation were separately defined based on the time at which the test or simulation exhibited dispersed flow film boiling heat transfer conditions characteristic of the reflood time period. The revised heat transfer multiplier distributions have been evaluated for impact on existing analyses. Resolution of these issues represents a closely related group of Non-Discretionary Changes in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Models

2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

A plant transient calculation representative of **AP1000®** transient behavior was performed with the latest version of WCOBRA/TRAC. Using this transient, a matrix of HOTSPOT calculations was performed to estimate the effect of the heat transfer multiplier distribution changes. Using these results and considering the heat transfer multiplier uncertainty attributes from limiting cases for **AP1000**, an estimated PCT effect of +11°F has been established for 10 CFR 50.46 reporting purposes for the **AP1000** Core Reference Report analysis.

2013 REPORTING

ERROR IN BURST STRAIN APPLICATION

Background

An error in the application of the burst strain was discovered in HOTSPOT. The equation for the application of the burst strain is given as Equation 7-69 in WCAP-16009-P-A and in WCAP-12945-P-A. The outer radius of the cladding after burst occurs should be calculated based on the burst strain, and the inner radius of the cladding should be calculated based on the outer radius. In HOTSPOT, the burst strain is applied to the calculation of the cladding inner radius. The cladding outer radius is then calculated based on the inner radius. As such, the burst strain is incorrectly applied to the inner radius rather than the outer radius, which impacts the resulting cladding geometry at the burst elevation after burst occurs. Correction of the erroneous calculation results in thinner cladding at the burst node and more fuel relocating into the burst node, leading to an increase in the Peak Cladding Temperature (PCT) at the burst node. This issue has been evaluated to estimate the impact on existing Best-Estimate (BE) Large-Break Loss-of-Coolant Accident (LBLOCA) analysis results. The resolution of this issue represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Models

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model

2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

The issue described above was evaluated by executing the most limiting plant-specific HOTSPOT runs with a HOTSPOT version that includes the correction of this error. This plant-specific sensitivity study resulted in an estimated PCT impact of 23°F for the Core Reference Report (CRR) analysis for the **AP1000®** plant.

2013 REPORTING

CHANGES TO GRID BLOCKAGE RATIO AND POROSITY

Background
A change in the methodology used to calculate grid blockage ratio and porosity for Westinghouse fuel resulted in a change to the grid inputs used in the **AP1000**® plant large break loss-of-coolant accident (LBLOCA) analysis. Grid inputs affect heat transfer in the core during a LBLOCA. This change represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)
2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect
The updates to the methodology to calculate grid blockage ratio and porosity used as input in Westinghouse LBLOCA models resulted in a negligible change to heat transfer in the core for the fuel type used in the **AP1000** plant Core Reference Report LBLOCA analysis. The estimated penalty associated with the changes is 0°F for 10 CFR 50.46 reporting purposes.

2013 REPORTING

INITIAL FUEL PELLET AVERAGE TEMPERATURE UNCERTAINTY CALCULATION

Background

In the Automated Statistical Treatment of Uncertainty Method (ASTRUM) Best-Estimate (BE) Large-Break Loss-of-Coolant Accident (LBLOCA) Evaluation Model (EM), uncertainties are applied to the gap heat transfer coefficient and pellet thermal conductivity to capture the uncertainty in the initial fuel pellet average temperature. This approach was compared to the initial fuel pellet average temperature uncertainties predicted by the PAD code at beginning-of-life conditions and found to be in agreement per Footnote 10 to Table 25-4-11 of WCAP-12945-P-A. However, the initial fuel pellet average temperature uncertainty range analyzed at higher burnups in the ASTRUM EM is inconsistent with the uncertainty range predicted by the PAD code. This issue has been evaluated to estimate the impact on existing ASTRUM LBLOCA analysis results. The resolution of this issue represents a Non-Discretionary Change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)

2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

The issue described above was evaluated with plant-specific sensitivity studies resulting in an estimated Peak Cladding Temperature (PCT) impact of 0°F.

2013 REPORTING

AUTOMATIC DEPRESSURIZATION SYSTEM STAGE 4 VALVE SINGLE FAILURE ASSUMPTION

Background
The failure of an automatic depressurization system stage 4 (ADS-4) valve is the limiting single failure for the **AP1000®** plant small break loss-of-coolant accident analyses. Historically, that failure has been assumed to be on the pressurizer side of the reactor coolant system (RCS). However, experimental results indicate that the failure on the non-pressurizer side of the RCS is the limiting failure. Analysis work was completed with the most recent **AP1000** plant SBLOCA analysis model to determine the impact of this change. The period of interest with respect to ADS-4 performance is the transition from core makeup tank (CMT) injection to in-containment refueling water storage tank (IRWST) injection. Reduced ADS-4 performance may cause a delay in the onset of IRWST injection, causing or extending a gap in the direct vessel injection (DVI) line make-up water injection.

This represents a Non-Discretionary Change to the Evaluation Model as described at Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)
1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
The limiting break size for SBLOCA in the **AP1000** plant design control document (DCD) is the 10 inch cold leg break with the peak cladding temperature (PCT) occurring very early in the transient. The 10 inch cold leg break analysis with the most recent model demonstrates that the change in failure assumption has little impact on the transient results for this break size since the break itself alleviates some of the demand on the ADS-4 system. The analysis of smaller breaks with the most recent model shows that while the impact of the single failure assumption change is larger, the resulting PCT is significantly lower than the PCT for the 10 inch cold leg break presented for the DCD. As such, the estimated effect of the change in single failure assumption on the PCT is 0°F for the DCD SBLOCA analysis.

The **AP1000** plant Core Reference Report SBLOCA analysis includes the correction of this error leading to a PCT impact of 0°F.

2013 REPORTING

CORE MAKEUP TANK DISCHARGE PIPING MODELING ERROR

Background
Since AP600 analyses, a failure in one of the core makeup tank (CMT) parallel discharge paths has been accounted for in the calculation of loss coefficients input into the **AP1000®** plant small break loss-of-coolant accident (SBLOCA) analysis NOTRUMP model. This is an additional failure and is incorrect as the limiting single failure assumption for **AP1000** plant SBLOCA analyses is the failure of an automatic depressurization system stage 4 (ADS-4) valve. When a failure in the CMT discharge piping is assumed, the higher resistance slows tank draining during injection and extends the CMT injection period. With the elimination of this failure, both CMT parallel discharge paths are open resulting in lower resistance, which causes higher CMT discharge flow rates and faster tank draining. An earlier emptying of the CMTs can lead to a gap or increase the duration of a gap between the end of CMT injection and the beginning of in-containment refueling water storage tank (IRWST) injection.

This represents a Non-Discretionary Change to the Evaluation Model as described in Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)
1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
The limiting break size for SBLOCA in the **AP1000** plant design control document (DCD) is the 10 inch cold leg break with the peak cladding temperature (PCT) occurring very early in the transient. Analysis work was completed with the most recent **AP1000** plant SBLOCA analysis model and demonstrates that the CMT discharge loss coefficient change results in slightly earlier ADS actuation times for the 10 inch break, but negligibly impacts the overall 10 inch cold leg break transient response. The updated analysis of smaller breaks shows that while this change has an impact, the resulting PCT is significantly lower than the PCT for the 10 inch cold leg break presented for the DCD. As such, the estimated effect of the change in CMT discharge piping modeling on the PCT is 0°F for the DCD SBLOCA analysis.

The **AP1000** plant Core Reference Report SBLOCA analysis includes the correction of this error leading to a PCT impact of 0°F.

2013 REPORTING

FLOAD4 TEE MODEL ERROR

Background
The FLOAD4 code calculation results supplies the automatic depressurization system stage 4 (ADS-4) loss coefficient adjustment factor for use in the **AP1000**® plant small break loss-of-coolant accident (SBLOCA) NOTRUMP model at the transition to non-critical flow. The calculation was updated to correct an error in the tee model after the **AP1000** plant design control document (DCD) analyses were completed. The result of the error correction is a larger adjustment factor than previously used for the DCD analyses.

This represents a Non-Discretionary Change to the Evaluation Model as described at Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)
1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
The increase in the loss coefficient adjustment factor impacts transient response only after the ADS-4 valves have opened and the flow paths have transitioned to non-critical flow. Thus, the currently limiting peak cladding temperature (PCT) for the 10 inch cold leg break is not directly impacted by this error as the PCT occurs before ADS-4 actuation. Analysis work was completed with the most recent **AP1000** plant SBLOCA analysis model to further assess the impact of this error. The simulation of the 10 inch cold leg break shows that the overall transient behavior following ADS-4 actuation is very similar when considering this error. The updated analysis simulations completed for smaller breaks indicate that the PCT remains far below the currently limiting PCT for the 10 inch cold leg break. As such, the estimated effect of the change in loss coefficient adjustment factor as a result of the FLOAD4 tee model error on the PCT is 0°F for the DCD SBLOCA analysis.

The **AP1000** plant Core Reference Report SBLOCA analysis includes the correction of this error leading to a PCT impact of 0°F.

2013 REPORTING

AUTOMATIC DEPRESSURIZATION SYSTEM STAGE-4 NON-CRITICAL LOSS COEFFICIENT ADJUSTMENT ERROR

Background
The FLOAD4 model considers the entire automatic depressurization system stage 4 (ADS-4) flow path from the hot leg to the ADS-4 valves. The loss coefficient adjustment calculations, for the ADS-4 path on the pressurizer side of the reactor coolant system, as performed in the **AP1000®** plant design control document (DCD) small break loss-of-coolant accident (SBLOCA) analyses only consider a portion of the piping for the ADS-4 path branching from the passive residual heat removal (PRHR) heat exchanger inlet piping rather than the piping starting from the hot leg. Correction of this error leads to a slightly higher loss coefficient for input into NOTRUMP after the ADS-4 flow paths have transitioned to non-critical flow

This represents a Non-Discretionary Change to the Evaluation Model as described at Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)
1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
The increase in loss coefficient for the affected path is relatively small and impacts transient response only after the ADS-4 valves have opened and the flow paths have transitioned to non-critical flow. Thus, the currently limiting peak cladding temperature (PCT) for the 10 inch cold leg break is not impacted by this error as the PCT occurs before ADS-4 actuation. Analysis work was completed with the most recent **AP1000** plant SBLOCA analysis model to further assess the impact of this error. The simulation of the 10 inch cold leg break shows that the overall transient behavior following ADS-4 actuation is very similar when considering this error. The updated analysis simulations completed for smaller breaks indicate that the PCT remains far below the currently limiting PCT for the 10 inch cold leg break. As such, the estimated effect of the change in the ADS-4 non-critical loss coefficient adjustment calculations on the PCT is 0°F for the DCD SBLOCA analysis.

The **AP1000** plant Core Reference Report SBLOCA analysis includes the correction of this error leading to a PCT impact of 0°F.

2013 REPORTING

ERROR IN PASSIVE RESIDUAL HEAT REMOVAL HEAT EXCHANGER/AUTOMATIC DEPRESSURIZATION SYSTEM STAGE-4 INLET PIPING LOSS COEFFICIENTS

Background
The **AP1000**® plant small break loss-of-coolant accident (SBLOCA) analysis model for the design control document (DCD) double accounted for loss coefficients in the passive residual heat removal (PRHR) heat exchanger inlet common piping with the automatic depressurization system stage 4 (ADS-4). As such, the loss coefficient for the ADS-4 piping path from the hot leg to the valves was modeled higher than designed for **AP1000** SBLOCA analysis results.

This represents a Non-Discretionary Change to the Evaluation Model as described at Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)
1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
With respect to the limiting 10 inch cold leg break, the error in resistance only occurs while ADS-4 is discharging reactor coolant system (RCS) inventory which is significantly after the peak cladding temperature (PCT) time for the 10 inch cold leg break. Analysis work was completed with the most recent **AP1000** plant SBLOCA analysis model to further assess the impact of this error correction. The simulation of the 10 inch cold leg break shows that the overall transient behavior following ADS-4 actuation is very similar when considering this error correction. The updated analysis simulations completed for smaller breaks indicate that the PCT remains far below the currently limiting PCT for the 10 inch cold leg break. As such, the estimated effect of the change in the PRHR heat exchanger/ADS-4 inlet piping loss coefficients on the PCT is 0°F for the DCD SBLOCA analysis.

The **AP1000** plant Core Reference Report SBLOCA analysis includes the correction of this error leading to a PCT impact of 0°F.

2013 REPORTING

SBLOCTA CLADDING STRAIN REQUIREMENT FOR FUEL ROD BURST

Background
An error was discovered in the minimum local strain required for burst for **ZIRLO®** cladding in the SBLOCTA code. The coding does not enforce reaching the minimum percent local strain threshold prior to calculating fuel rod burst. However, a review of licensing basis analyses revealed no instances of this error impacting calculated results. Resolution of this issue represents a Non-Discretionary Change to the Evaluation Model as described in Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)
1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
The SBLOCTA code is not used for the **AP1000®** Design Control Document (DCD) small break LOCA analysis. The SBLOCTA code is used for the **AP1000** plant Core Reference Report (CRR) analysis but no fuel rod burst is calculated to occur. As a result, the estimated peak cladding temperature (PCT) impact is 0°F.

2013 REPORTING

GENERAL CODE MAINTENANCE

Background
Various changes have been made to enhance the usability of the codes and to help preclude errors in analyses. This includes items such as modifying input variable definitions, units, and defaults; improving the input diagnostic checks; enhancing the code output; optimizing active coding; and, eliminating inactive coding. These changes represent Discretionary Changes that will be implemented on a forward-fit basis in accordance with Section 4.1.1 of WCAP-13451.

Affected Evaluation Model(s)
1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
The nature of these changes leads to an estimated PCT impact of 0°F.

2013 REPORTING

CORE SHROUD MODELING UPDATE

Background
The **AP1000**® plant core barrel-shroud region of the reactor vessel is modeled in a simplified manner for the NOTRUMP-AP600 evaluation model. Flow communication into and out of the region is not modeled. The volume of this region is no longer being accounted for, consistent with the approach to the flow communication modeling and to ensure model conservatism.

This change represents a Discretionary Change that will be implemented on a forward-fit basis in accordance with Section 4.1.1 of WCAP-13451.

Affected Evaluation Model(s)
1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
This change in core barrel-shroud volume modeling has the potential to provide a more realistic prediction of thermal hydraulic behavior in the active fuel region but overall is expected to have little impact on transient behavior. Past transient simulations performed with this change and various design changes and the **AP1000** plant Core Reference Report (CRR) simulations confirm that the impact is minimal. As a result, the estimated PCT impact is 0°F.

2013 REPORTING

INADVERTENT ADS TRANSIENT SPARGER MODELING UPDATE

Background
For the **AP1000®** plant inadvertent automatic depressurization system (ADS) transient simulations the discharge from the ADS 1-3 flow path is choked for only a short period of time. Historically, for AP600 plant analyses the flow path type was changed from a "break" flow path to a "normal" flow path in this process to improve computational performance. As a simplification **AP1000** plant analyses did not make this flow path change. For the **AP1000** plant Core Reference Report (CRR) analyses and moving forward the flow path is modeled as a "break" flow path and changed to a "normal" flow path when warranted.

This change represents a Discretionary Change that will be implemented on a forward-fit basis in accordance with Section 4.1.1 of WCAP-13451.

Affected Evaluation Model(s)
1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
Based upon multiple transient simulations and the **AP1000** plant CRR analysis this change has a negligible impact on results and mostly serves to improve computational performance. As such, the estimated PCT impact is 0°F.

2013 REPORTING

PRHR HEAT EXCHANGER OUTLET PIPING INITIAL TEMPERATURE

Background
The passive residual heat removal (PRHR) heat exchanger outlet piping in the **AP1000®** plant small break LOCA NOTRUMP model was previously set to a generic temperature and is now being modeled at a temperature more consistent with the plant design and analyses.

This change represents a Discretionary Change that will be implemented on a forward-fit basis in accordance with Section 4.1.1 of WCAP-13451.

Affected Evaluation Model(s)
1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
This change only impacts the PRHR outlet piping initial temperature; transient simulations in the **AP1000** plant Core Reference Report (CRR) confirm that this change has a negligible impact on the transient results. As a result, the estimated PCT impact is 0°F.

2013 REPORTING

IRWST VENT AND OVERFLOW MODELING

Background
The in-containment refueling water storage tank (IRWST) in the **AP1000®** plant small break LOCA NOTRUMP model has historically only contained a generic tank overflow and no tank vents. As a result, the IRWST was observed to experience unrealistic pressure variations. In order to more realistically capture the IRWST tank behavior during a small break LOCA transient, a vent path has been added to the model and the generic overflow data has been replaced with **AP1000** plant specific data.

This change represents a Discretionary Change that will be implemented on a forward-fit basis in accordance with Section 4.1.1 of WCAP-13451.

Affected Evaluation Model(s)
1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
This change prevents unrealistic pressure variations in the IRWST. This change has a negligible impact on the overall transient results as confirmed in the **AP1000** plant Core Reference Report (CRR) simulations. As a result, the estimated PCT impact is 0°F.

2013 REPORTING

IRWST CHECK VALVE CRACKING/OPENING PRESSURE UPDATE

Background

Historically, a generic value for the cracking/opening pressure of the check valves between the in-containment refueling water storage tank (IRWST) and direct vessel injection (DVI) line was used in the **AP1000**® plant small break LOCA analyses. It was recently determined that the generic value was not supported by the current **AP1000** plant design and as such a higher check valve cracking pressure reflective of the plant design will be used.

This change represents a Discretionary Change that will be implemented on a forward-fit basis in accordance with Section 4.1.1 of WCAP-13451.

Affected Evaluation Model(s)

1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect

This forward fit change was implemented in the **AP1000** plant Core Reference Report (CRR) analysis leading to an estimated PCT impact is 0°F.

2013 REPORTING

DIRECT VESSEL INJECTION LINE MODELING

Background
The direct vessel injection (DVI) line model in the **AP1000®** plant small break LOCA NOTRUMP model has been updated to allow for a more physical representation of the piping and more closely reflect the modeling used for the integral effects test (IET) validation for the **AP1000** plant. This update includes modeling fluid node boundaries at physical pressure boundaries such as check valves and updating input parameters to ensure appropriate behavior.

This change represents a Discretionary Change that will be implemented on a forward-fit basis in accordance with Section 4.1.1 of WCAP-13451.

Affected Evaluation Model(s)
1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
This forward fit change was implemented in the **AP1000** plant Core Reference Report (CRR) analysis leading to an estimated PCT impact is 0°F.

2013 REPORTING

STEAM GENERATOR OUTLET PLENUM AND REACTOR COOLANT PUMP NOZZLE MODELING UPDATE

Background

The fluid nodes and flow paths representing the steam generator outlet plenum and reactor coolant pump (RCP) inlet nozzle in the **AP1000**® plant small break LOCA NOTRUMP model have been updated to allow for a more realistic representation of the **AP1000** plant design. This update better captures the inside bottom of the steam generator outlet plenum and the connection with the RCP inlet nozzles.

This change represents a Discretionary Change that will be implemented on a forward-fit basis in accordance with Section 4.1.1 of WCAP-13451.

Affected Evaluation Model(s)

1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect

This forward fit change was implemented in the **AP1000** plant Core Reference Report (CRR) analysis leading to an estimated PCT impact is 0°F.

2013 REPORTING

UNAPPROVED DRAWING USED IN AP1000 PLANT CORE REFERENCE REPORT SMALL BREAK LOCA MODEL DEVELOPMENT

Background
It was discovered that an unapproved piping drawing related to the Automatic Depressurization System (ADS) Stages 1-3 was used during the **AP1000**® plant small break LOCA model development for the Core Reference Report (CRR).

Resolution of this issue represents a Non-Discretionary Change to the Evaluation Model as described in Section 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)
1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
Based upon review of an approved version of the drawing, the resulting input changes were judged to have a negligible impact on the small break LOCA analysis results. As such, the estimated peak cladding temperature (PCT) impact is 0°F for the **AP1000** plant CRR analysis.

2013 REPORTING

CODE UPDATE FOR THE APPLICATION OF TIME DEPENDENT CONTAINMENT BACKPRESSURE IN THE AP1000 PLANT CORE REFERENCE REPORT SMALL BREAK LOCA ANALYSIS

Background
An updated version of the NOTRUMP-AP600 code was used for certain small break LOCA analysis transient simulations presented in Revision 1 of the **AP1000®** plant Core Reference Report (CRR). This code version allows for the use of a time dependent containment backpressure curve as input. The use of this input is discussed in Revision 1 of the **AP1000** plant Core Reference Report (CRR).

This change represents a Discretionary Change that is implemented for the CRR SBLOCA analysis in accordance with Section 4.1.1 of WCAP-13451.

Affected Evaluation Model(s)
1985 Westinghouse Advanced Plant Small Break LOCA Evaluation Model with NOTRUMP

Estimated Effect
This change was implemented in the **AP1000** plant Core Reference Report (CRR) analysis and is reflected in the results where applicable.

Westinghouse LOCA Peak Clad Temperature Summary for ASTRUM Best Estimate Large Break

Future

Plant Name: AP1000
Utility Name: Westinghouse Nuclear Power Plants
Revision Date: 3/17/2014

Analysis Information

EM:	ASTRUM (2004)	**Analysis Date:**	12/11/2012	**Limiting Break Size:**	DEG
FQ:	2.6	**FdH:**	1.72		
Fuel:	17x17 AP1000	**SGTP (%):**	10		

Notes: Plant specific adaptation of the ASTRUM EM which explicitly accounts for effects of thermal conductivity degradation and peaking factor burndown.

	Clad Temp (°F)	Ref.	Notes
LICENSING BASIS			
Analysis-Of-Record PCT	1936	1	(a)
PCT ASSESSMENTS (Delta PCT)			
A. PRIOR ECCS MODEL ASSESSMENTS			
1 . None	0		
B. PLANNED PLANT MODIFICATION EVALUATIONS			
1 . None	0		
C. 2013 ECCS MODEL ASSESSMENTS			
1 . Revised Heat Transfer Multiplier Distributions	11	2	
2 . Error in Burst Strain Application	23	3	
D. OTHER*			
1 . None	0		
LICENSING BASIS PCT + PCT ASSESSMENTS PCT =	1970		

* It is recommended that the licensee determine if these PCT allocations should be considered with respect to 10 CFR 50.46 reporting requirements.

References

1 . LTR-NRC-12-86, "Westinghouse Response to NRC RAIs on WCAP-17524, "AP1000 Core Reference Report," (Proprietary / Non-Proprietary)," January 2, 2013.

2 . LTR-LIS-13-357, "AP1000 Plant 10 CFR 50.46 Report for Revised Heat Transfer Multiplier Distributions," July 2013.

3 . LTR-LIS-14-41, "AP1000 Plant 10 CFR 50.46 Report for the HOTSPOT Burst Strain Error Correction," January 2014.

Notes:

(a) Value contains 2°F bias for PCT sensitivity to PRHR isolation, per Reference 1 response to CRR-008, Table 2 and Table 15.6.5-8.

Westinghouse LOCA Peak Clad Temperature Summary for Appendix K Small Break

Future

Plant Name: AP1000
Utility Name: Westinghouse Nuclear Power Plants
Revision Date: 2/27/2014

Analysis Information

EM:	NOTRUMP-AP	**Analysis Date:**	8/26/2013	**Limiting Break Size:**	2 Inch
FQ:	2.6	**FdH:**	1.75		
Fuel:	RFA	**SGTP (%):**	10		
Notes:					

	Clad Temp (°F)	Ref.	Notes
LICENSING BASIS			
Analysis-Of-Record PCT	663.5	1	
PCT ASSESSMENTS (Delta PCT)			
A. PRIOR ECCS MODEL ASSESSMENTS			
1 . None	0		
B. PLANNED PLANT MODIFICATION EVALUATIONS			
1 . None	0		
C. 2013 ECCS MODEL ASSESSMENTS			
1 . None	0		
D. OTHER*			
1 . None	0		
LICENSING BASIS PCT + PCT ASSESSMENTS **PCT =**	663.5		

* It is recommended that the licensee determine if these PCT allocations should be considered with respect to 10 CFR 50.46 reporting requirements.

References

1 . LTR-LIS-13-459, "10 CFR 50.46 Report for the AP1000 Plant Core Reference Report (CRR) Small Break LOCA (SBLOCA) Analysis," October 2013.

Notes:

(a) None

Section L

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LTR-NRC-14-25

May 1, 2014

Subject: Submittal of "Supplemental Information to Correct a Small Break LOCA Related Statement in WCAP-17524, Revision 1" (Non-Proprietary)

Subsequent to the submittal of WCAP-17524, Revision 1, "AP1000 Core Reference Report," an inconsistency was identified in the text. This inconsistent text does not affect the results or conclusions presented in WCAP-17524, Revision 1. To address this issue, the proposed change has been described in the attachment and will be corrected during preparation of the approved version. Attached is the non-proprietary version of, "Supplemental Information to Correct a Small Break LOCA Related Statement in WCAP-17524, Revision 1."

If you have any questions or require additional information, please contact Keith Drudy at (412) 374-5841.

Very truly yours,

James A. Gresham, Manager
Regulatory Compliance

COPYRIGHT NOTICE

Supplemental Information to Correct a Small Break LOCA Related Statement in WCAP-17524, Revision 1 (Non-Proprietary)

May 2014

Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, PA 16066

Supplemental Information to Correct a Small Break LOCA Related Statement in WCAP-17524, Revision 1

A revision to the **AP1000**® Pressurized Water Reactor Core Reference Report (Reference 1) was transmitted to the NRC by letter dated March 21, 2014. Revision 1 was created to address changes and errors in the analyses presented in Revision 0. As part of Revision 1 the Small Break LOCA (SBLOCA) analysis was updated. Subsequent to the submittal of WCAP-17524, Revision 1, an inconsistency was identified in the text of the revised SBLOCA section. The incorrect text in Section 15.6.5.4B.2.1, Page 15.6-38 (last paragraph) reads as follows:

"The small-break LOCA spectrum analyzed for **AP1000** includes breaks that exhibit a minimum reactor vessel inventory early in the transient, before the accumulators become active: the DEDVI and 10-inch cold leg break."

The results of the 10-inch break transient documented in Section 15.6.5.4B.3.6 indicate that the minimum reactor vessel inventory does not occur early in the transient, but later in the transient closer to when the in-containment refueling water storage tank (IRWST) injection begins (see Reference 1, Figure 15.6.5.4B-68(b)). The statement still applies to the DEDVI break at 20 psia containment backpressure for which the minimum reactor vessel inventory occurs early in the transient, closer to when accumulator injection begins (see Reference 1, Figure 15.6.5.4B-53(b)). The corrected statement should read as follows:

"The small-break LOCA spectrum analyzed for **AP1000** includes a break that exhibits a minimum reactor vessel inventory early in the transient, before the accumulators become active: the DEDVI break at 20 psia containment backpressure."

This correction is minor and does not affect the overall results and conclusions documented in the report. This proposed change will be implemented in the creation of the approved version of Reference 1. Additionally, a page markup has been provided as Attachment 1.

Reference

1. WCAP-17524-P/WCAP-17524-NP, Revision 1, "**AP1000** Core Reference Report," March 2014.

Attachment 1

isolation valves are ramped closed between 2 and 7 seconds after the “S” signal. The reactor coolant pumps are tripped 7.3 seconds after the “S” signal.

- The ADS actuation signals are generated on low core makeup tank levels and the ADS timer delays. A list of the ADS parameters is given in Table 15.6.5-10 for AP1000. ADS Stages 1, 2, and 3 are modeled as discharging through spargers submerged in the IRWST at the appropriate depth.

- The Inadvertent ADS actuation and 2-inch cold leg break NOTRUMP simulations utilize a time-dependent containment pressure in the boundary node modeling of the containment. These conditions were generated by providing mass and energy releases from these AP1000 breaks to the AP1000 WGOTHIC containment model while the WGOTHIC code calculates the containment pressure response. The Inadvertent ADS actuation and 2-inch cold leg break NOTRUMP simulations then utilized the time-dependent pressure history curves as generated by WGOTHIC. The 10-inch cold leg break case models a pressure in the boundary node of the containment of 14.7 psia and the DEDVI line break models two cases with a constant 20 psia and 14.7 psia containment backpressure, respectively. The steam generator secondary is isolated 6 seconds after the reactor trip signal, due to closure of the turbine stop valves. The main steam safety valves actuate and remove energy from the steam generator secondary when pressure reaches 1235 psia.

Active single failures of the passive safeguards systems are considered. The limiting failure is judged to be one out of four ADS Stage 4 valves failing to open on demand, the failure that most severely impacts depressurization capability. The safety design approach of the AP1000 is to depressurize the reactor coolant system to the containment pressure in an orderly fashion such that the large reservoir of water stored in the IRWST is available for core cooling. The mass inventory plots provided for the breaks show the minimum inventory condition generally occurs at the start of IRWST injection. Penalizing the depressurization is the most conservative approach in postulating the single failure for such breaks.

The small-break LOCA spectrum analyzed for AP1000 includes a break that exhibits a minimum reactor vessel inventory early in the transient, before the accumulators become active: the DEDVI break at 20 psia containment backpressure. In this transient, the early mass inventory decrease is terminated by injection flow from the accumulators, and depressurization through the break enables accumulator injection to begin with no contribution from the actuation of ADS Stages 1, 2, and 3. For consistency, the conservative failure of one of the ADS Stage 4 valves located off the non-pressurizer loop, which adversely affects the depressurization necessary

Deleted: s

Deleted: and 10-inch cold leg

Section M

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LTR-NRC-14-32

June 13, 2014

Subject: Supplemental Information to Correct a Typographical Error Related to the RCCA Insertion Time in WCAP-17524, Revision 1 "**AP1000** Core Reference Report" (Non-Proprietary)

Reference: 1) WCAP-17524, Revision 1, "**AP1000** Core Reference Report" March 2014

Subsequent to the submittal of WCAP-17524, Revision 1, "**AP1000** Core Reference Report," (Reference 1) an inconsistency was identified in the text. This inconsistent text does not affect the results or conclusions presented in Reference 1. To address this issue, the proposed change has been described in the attachment and will be corrected during preparation of the approved version. Attached is the non-proprietary version of, "Supplemental Information to Correct a Typographical Error Related to the RCCA Insertion Time in WCAP-17524, Revision 1 '**AP1000** Core Reference Report' (Non-Proprietary)."

If you have any questions or require additional information, please contact Keith Drudy at (412) 374-5841.

Very truly yours,

J A Gresham

James A. Gresham, Manager
Regulatory Compliance

Supplemental Information to Correct a Typographical Error Related to the RCCA Insertion Time in WCAP-17524 Revision 1, "*AP1000* Core Reference Report" (Non-Proprietary)

June 2014

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Supplemental Information to Correct a Typographical Error Related to the RCCA Insertion Time in WCAP-17524, Revision 1 "*AP1000* Core Reference Report"

A revision to the **AP1000**® Pressurized Water Reactor Core Reference Report (Reference 1) was transmitted to the NRC by letter dated March 21, 2014. Revision 1 was created to address changes and errors in the analyses presented in Revision 0. Subsequent to the submittal of Reference 1, an inconsistency was identified in the description of an assumption in the rod drop time. In section 15.3.1.2.3, the rod drop time used for the partial loss of forced reactor coolant flow analysis is stated as 2.3 seconds; however, in section 15.0.5, it is stated:

"In analyses where some or all of the reactor coolant pumps are running, the RCCA insertion time to dashpot is conservatively taken as 2.7 seconds."

Since some reactor coolant pumps continue to operate, the 2.7 second time presented in 15.0.5 is applicable to the partial loss of forced reactor coolant flow accident, and the time presented in 15.3.1.2.3 is in error. The analysis addressed in Section 15.3.1 was performed using the correct time of 2.7 seconds; therefore, the correction of this error does not result in any changes to the results and conclusions presented in Reference 1.

This proposed change will be implemented in the creation of the approved version of Reference 1. Additionally, a page markup has been provided as Attachment 1 to this transmittal.

Reference

1) WCAP-17524, Revision 1, "**AP1000** Core Reference Report" March 2014

Attachment 1

15.3.1.2.3 Reactivity Coefficients

The reactivity feedback parameters are chosen so as to maximize the energy transferred to the primary coolant during the flow coastdown. A most-negative Doppler-only power coefficient (see Figure 15.0.4-1) is applied to maximize the positive reactivity addition during the reactor trip and rod motion, which acts to slow the rate of power reduction; the equivalent total integrated Doppler reactivity from 0 to 100 percent power of 0.016 Δk. As there is an initial heatup due to the reduction in RCS flow, a least-negative (minimum feedback) moderator temperature coefficient is most conservative. Therefore, a constant moderator density coefficient of 0.0 Δk/g/cc is modeled. Finally, a curve of trip reactivity versus time based on a ~~2.3~~ 2.7-second rod cluster control assembly insertion time to the dashpot is applied (see subsection 15.0.5).

15.3.1.2.4 Flow Coastdowns

Conservative flow coastdowns are used to simulate the transient. The flow coastdowns are calculated externally to the LOFTRAN code using the COAST computer code which is described in Section 15.0.11.

15.3.1.2.5 Protection Systems

Plant systems and equipment necessary to mitigate the effects of the accident are discussed in subsection 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or equipment adversely affects the consequences of the accident.

15.3.1.2.6 Results

Figures 15.3.1-1 through 15.3.1-6 show the transient response for the loss of two reactor coolant pumps with offsite power available. Figure 15.3.1-6 demonstrates that the DNBR is always greater than the safety analysis limit value, which demonstrates that the DNB design basis is met. The DNB design basis is described in Section 4.4.

The affected reactor coolant pumps coast down and the core flow reaches a new equilibrium value. The plant is tripped by the low-flow trip rapidly enough so that the capability of the reactor coolant to remove heat from the fuel rods is not greatly reduced. The average fuel and cladding temperatures do not increase significantly above their initial values. With the reactor tripped, a stable plant condition is attained and plant shutdown may then proceed.

The calculated sequence of events for the case analyzed is shown in Table 15.3-1.

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LTR-NRC-14-75

November 17, 2014

Subject: Westinghouse Response to NRC RAI Letter No. 3 on WCAP-17524, Revision 1, "AP1000 Core Reference Report" (Non-Proprietary)

By letter dated October 15, 2014 the NRC issued Request for Additional Information (RAI) letter No. 3 for the review of Westinghouse topical report WCAP-17524, Revision 1 "AP1000 Core Reference Report." Enclosed is the non-proprietary version of the Westinghouse response to the RAI, "Westinghouse Response to NRC RAI Letter No. 3 on WCAP-17524, Revision 1, 'AP1000 Core Reference Report.'"

If you have any questions or require additional information, please contact Keith Drudy at (412) 374-5841.

Very truly yours,

J. A. Gresham

James A. Gresham, Manager
Regulatory Compliance

Enclosures

cc: Bruce Bavol, NRC

Westinghouse Response to NRC RAI Letter No. 3 on WCAP-17524, Revision 1, "AP1000 Core Reference Report" (Non-Proprietary)

November 2014

Westinghouse Electric Company
1000 Westinghouse Drive
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CRR-029

In the Chapter 15 FSAR mark-ups found in Appendix B of WCAP-17524-P, several design basis accidents (e.g. 15.2.7, 15.5.1, 15.5.2, 15.4.6, and 15.6.2) now include descriptions of required operator actions that were not listed in the certified design. Section 15.0.13 contains a high level description of the operator actions and the associated change roadmap (page B-3 of WCAP-17524-P) summarizes the change as being consistent with the Westinghouse response to RAI-SRP15.0-SRSB-03, Rev. 2 from the DCD Rev. 19 review. The change roadmap for each of the affected design basis accidents identifies the operator actions as editorial changes. The staff notes that the response for SRP15.0-SRSB-03 Rev. 2 states that only Chapter 15.6.2 ("Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment") assumes operator action.

a) Are these required operator actions new to the core reference report or are they part of the DCD licensing basis? Describe the actions, including references to DCD documentation if applicable, and whether they are credited in the safety analyses intended to be included in the licensing basis for COL applications based on the DCD.

b) Explain how the Chapter 15 FSAR changes listed in Appendix B of the core reference report support the Section 6.3.4 statement that "the passive core cooling system can maintain safe shutdown conditions for 72 hours after an event without operator action and without both nonsafety-related onsite and offsite power".

Westinghouse Response to CRR-029

Part a

The operator actions presented in the Core Reference Report (CRR) WCAP-17524-P, Revision 1 are consistent with those previously presented in Revision 19 of the Design Control Document (DCD) for the **AP1000**® Pressurized Water Reactor (PWR), as well as prior revisions of the AP600 and **AP1000** PWR DCDs. This history is included in the staff's assessment of the AP600 and **AP1000** designs as presented in their associated Safety Evaluation Reports (SER). The following excerpt is from Section 15.2.5.1 of the SER for the AP600 PWR (NUREG-1512):

> *In an NRC quality control inspection at Westinghouse from November 17 through 21, 1997, the staff found that in some scenarios, operator actions are necessary (opening of the reactor vessel head vents) to prevent the pressurizer from overfilling with water. However, the applicant had not provided a technical specification or an ITAAC for the reactor vessel head vents (RVHV) to ensure that they will reliably function as assumed in the design basis analysis. In Westinghouse letter DCP/NRC-1248, dated February 6, 1998, the applicant's response to RAI 440.753F proposed limiting conditions for operation in TS 3.4.17 to require that the RVHVs be operable for Modes 1 through 3. In addition, the TS LCO is applicable in Mode 4 when the RCS is not being cooled by the normal decay heat removal system (RNS). The surveillance requirements are specified to be consistent with the inservice testing program. The staff has reviewed the proposed TS and found that the LCOs, required actions, and surveillance requirements are consistent with a typical TS for the RVHVs.*

This discussion is continued in Section 5.4.12.1 of the SER for the **AP1000** PWR (NUREG-1793):

> *The RVHVS is primarily used during plant startup to properly vent air from the RV head and to fill the RCS. The RVHVS valves also provide an emergency letdown path with a letdown flow rate within the capabilities of the normal makeup system to prevent pressurizer overfill following long-term loss of heat sink events....In addition to the normal venting procedures during startup, the AP1000 RVHVS could also be used under a design-basis accident scenario. During an accident, the AP1000 design relies on the passive safety-related systems, such as the PRHR HX, to provide the safety-related function of core cooling. Therefore, the design does not require the SG U-tubes to be vented to provide coolability of the core. However, the RVHVS is used under loss of heat sink events where the pressurizer level can increase and eventually become water solid, following long-term operation of the CMTs. To avoid this occurrence, the functional restoration guidelines for a high pressurizer level in the ERG requires that the RV vent flow be established to provide a bleed path, in response to high-pressurizer level conditions, to reduce the RCS inventory and prevent pressurizer overfill. When the pressurizer level is sufficiently reduced, the operator recloses the head vent valves. In this case, the operator uses pressurizer level as the primary indication to control operation of the RV head vent.*

The head vent operator action is also discussed by the staff in the SER for the **AP1000** PWR in relation to the Tech Specs:

> *The purpose of TS 3.4.16 is to ensure operability of the manually operated RVHVs so that the control room staff can open them to prevent overfilling of the pressurizer during RCS coolant addition transients.*

Both the DCD Rev. 19 and CRR analyses require the following operator actions to meet Chapter 15 acceptance criteria:

- Isolation of sample line break
  - Credited in Section 15.6.2, "Failure of Small Lines Carrying Primary Coolant Outside Containment"
- Isolation of dilution sources
  - Credited in Section 15.4.6, "Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant System"; specifically, for a subset of cases analyzed for Modes 1 and 2
- Opening of the reactor vessel head vent valve to prevent event propagation as a result of pressurizer filling
  - Described in Sections 15.5.1, "Inadvertent Operation of the Core Makeup Tanks During Power Operation" and 15.5.2, "Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory"

For the first two actions listed above, the manner in which these operator actions are presented in Appendix B of WCAP-17524-P is the same as that presented in the corresponding section of DCD Rev. 19. In Section 15.6.2, operator action is credited to isolate a sample line break within 30 minutes to meet dose limit criteria. In the DCD Rev. 19 analysis of the "Chemical and Volume Control System

Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant System" event (Section 15.4.6), some of the cases analyzed required operator actions. An example is the case modeling Mode 1 operation with automatic rod control. The DCD Rev. 19 analysis demonstrates that there is sufficient time for operator actions to occur to prevent a loss of shutdown margin.

The third operator action credited in Chapter 15 of DCD Rev. 19 prevents pressurizer overfill by opening the reactor vessel head vent valve. This operator action was discussed in the text for both the inadvertent actuation of a core makeup tank (Section 15.5.1) and the chemical and volume control system malfunction that increases reactor coolant inventory (Section 15.5.2) events. For each of these events in DCD Rev. 19, the case presented was one that is more limiting (i.e., reaches a higher peak pressurizer water volume) than the cases that required operator action. For instance, Section 15.5.1.3 of DCD Rev. 19 describes the operator actions as follows:

> *For such events, the operator would take action to reduce the increase in coolant inventory. As the pressurizer water level would increase above the high pressurizer water level that normally isolates chemical and volume control system makeup, the normal letdown line could be placed into service to reduce the increase in coolant inventory. If letdown could not be placed into service, the operator could use the safety related reactor vessel head vent valves to reduce the increase in coolant inventory. For these events, following the procedures outlined in the Emergency Response Guidelines AFR-I.1, there is sufficient time for the operator to mitigate the consequences of this event, and the results of such an event have a greater margin to pressurizer overfill than that presented in this analysis.*

Based in part on the informal feedback from the staff during the review of Revision 17 of the DCD and lessons learned identified via Westinghouse's corrective action program, the CRR presents the limiting cases for the analyses in DCD Section 15.5.1 and 15.5.2 as those with the minimum time requirement for operator action instead of the case with the highest final pressurizer water volume. Furthermore, the CRR provides additional details describing the operator action to open the reactor vessel head vent valve and ensures that these operator actions are consistent with the symptoms of the event seen by the operators. Therefore, in both the DCD Revision 19 and CRR analyses for Sections 15.5.1 and 15.5.2, operator actions are required to show that the pressurizer does not reach a water solid condition; the difference is the manner in which they are presented.

While no additional operator actions are added in the CRR analyses, one additional event, the "Loss of Normal Feedwater Flow" event (Section 15.2.7), credits the same operator action modeled in Section 15.5.1 and 15.5.2: the opening of the reactor vessel head vent valve. The more limiting results are primarily due to the inclusion of the effects of containment back pressure on the Passive Residual Heat Removal Heat Exchanger (PRHR HX) operation which results in a reduction in the PRHR HX heat transfer relative to the DCD Rev. 19 analysis. For the CRR, the loss of normal feedwater (Section 15.2.7), inadvertent actuation of a core makeup tank (Section 15.5.1), and chemical and volume control system malfunction (Section 15.5.2) analyses, the credited operator action is based upon the high-2 pressurizer water level setpoint providing indication to the operator that a potential over-filling event is underway.

Part b

The changes to the text pertaining to required operator actions in the CRR are consistent with those required for DCD Rev.19 as shown in response to part (a) above. As such, the relationship between Section 6.3.4 and Chapter 15 is unchanged. Section 6.3.4, "Post-72 hour Actions," is discussing the ability of the PXS to maintain water supply without operator action in order to satisfy the success criteria associated with adequate core cooling. This text is not meant to imply that no operator actions are required to satisfy the more restrictive Chapter 15 acceptance criteria. Therefore, the required operator actions in Chapter 15 of the licensing basis do not conflict with the statement quoted from Section 6.3.4.

Section O

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 7, 2013

Mr. J. A. Gresham, Manager
Westinghouse Electric Company
Nuclear Services
1000 Westinghouse Drive
Cranberry Township, PA 16066

SUBJECT: AUDIT SUMMARY FOR REVIEW OF TOPICAL REPORT WCAP-17524, "AP-1000 CORE REFERENCE REPORT" AND SUPPLEMENTAL INFORMATION

Dear Mr. Gresham:

From July through September 2012, U.S. Nuclear Regulatory Commission (NRC) staff conducted a multiple phase audit related to the review of the Westinghouse topical report, WCAP-17524, "AP1000 Core Reference Report" and supplemental information. The enclosed audit summary describes each phase of the review, identifies the audit participants, and lists the questions that led to requests for additional information.

The audit took place in three phases, as previously discussed and was coordinated with Westinghouse. Phase 1 occurred July 31 – August 1, 2012, Phase 2 occurred August 14 - 16, 2012, and Phase 3 occurred September 11-12, 2012.

Prior to placing the non-proprietary version of the audit summary in the public document room the staff requests that Westinghouse perform a final review for proprietary information. If after a 5 day period, you do not request additional portions be withheld, the non-proprietary version of the audit summary will be made publically available.

If you have any questions, please contact the Project Manager, Bruce Bavol, at (301) 415-6715 or Bruce.Bavol@NRC.gov.

Sincerely,

Larry Burkhart, Acting Chief
Licensing Branch 4
Division of New Reactor Licensing
Office of New Reactors

Project No. 0793

Enclosures:
1. Audit Participation
2. Audit Summary – (Non-Proprietary)
3. Audit Summary – (Proprietary)

cc: See next page (w/o Enclosure 3)

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Audit Participation - July 31-August 1, August 14-16, and September 11-12, 2012

The following table provides a comprehensive list of the audit participants from the NRC, WEC, and applicable licensees.

Name	Affiliation	Date(s) Attended
Bruce Bavol	NRC	July, Aug, Sept, 2012
Anthony Minarik	NRC	July 2012
Chris Van Wert	NRC	July, Aug, Sept, 2012
Y. Gene Hsii	NRC	July, Aug, Sept, 2012
John Budzynski	NRC	July, Aug, Sept, 2012
Shanlai Lu	NRC	July, Aug, Sept, 2012
Jen Gall	NRC	July 2012
Ben Parks	NRC	July 2012
Amrit Patel	NRC	July, Aug, Sept, 2012
Kristopher Cummings	Westinghouse	July, Aug, Sept, 2012
Ryan Lenahan	Westinghouse	July, Aug, Sept, 2012
Jennifer Baker	SNC	July, Aug, Sept, 2012
Dave Huegel	Westinghouse	July, Aug, 2012
Paul Kersting	Westinghouse	July 2012
Keith Drudy	Westinghouse	July, Sept, 2012
Terry Schulz	Westinghouse	August 2012
Kevin McNamee	Westinghouse	August 2012
Tim Crede	Westinghouse	August 2012
Dan Golden	Westinghouse	August 2012
Mike Misvel	Westinghouse	September 2012
Ron Knott	Westinghouse	September 2012
John DeBlasio	Westinghouse	September 2012

Enclosure 1

Audit Summary for Reviewing WCAP-17524, "AP1000 Core Reference Report" and Supplemental Information
Westinghouse Electric Company
Project No. 0793
July 31-August 1, August 14-16, and September 11-12, 2012

Phase 1 Audit - WCAP-17524-P, "AP1000 Core Reference Report"

On July 31 and August 1, 2012 staff of the U.S. Nuclear Regulatory Commission (NRC) conducted Phase 1 of the audit at the Westinghouse Twinbrook Office in Rockville, MD. This phase of the audit focused on calculations related to GSI-191, Thermal Conductivity Degradation (TCD), and changes to the fuel design.

Pertaining to GSI-191, the staff heard presentations from Westinghouse personnel and reviewed applicable reference documents. After gathering information from interviews and document review the staff asked if Westinghouse could better explain and present their process of handling GSI-191 requirements during the next phase of the audit. Additionally, the staff requested Westinghouse provide information regarding specific changes to the midgrid spacer of the fuel assembly and to better explain what effects the design changes have on hydraulic performance.

Regarding TCD the staff had several questions and takeaways from this phase of the audit (a full list of audit requests for additional information (RAIs) are provided at the end of the summary). Westinghouse provided the staff with a presentation on the subject of TCD along with supporting technical documentation. The staff had a question on how Westinghouse chose to accommodate the effects of TCD, the focus being on the method used to calculate fission gas release fractions, and the accompanying model. Westinghouse agreed to address this concern during phase 2 of the audit. Second, the staff asked Westinghouse if they had considered the effects TCD has on average core temperature in its large break loss-of-coolant accident (LOCA) analysis, or if they performed a calculation assuming a hot rod or bundle. In this regard, the staff also requested Westinghouse provide the results from all 124 accident cases, rather than the 31 they chose to sample for determining the bounding accident when TCD is considered. Additionally, in Westinghouse's supplement to the CRR topical report, the FQ percent margin was reduced and Westinghouse offered to explain the reasoning in more detail during the next phase of the Audit.

Some additional information that Westinghouse planned to provide during phase 2 of the audit included; the reason for changing a pump in some accident analyses, whether the STAV code used by Westinghouse correctly matched with the Halden data on TCD, and the development of how the data was correlated. Finally the staff requested a simple walkthrough of the calculations, including both the particulars of certain sets run in Westinghouse document APP-SSAR-GSC-772 and the calculation used to account for peaking factor and associated burndown effects.

Enclosure 2

Phase 2 Audit

On August 14-16, 2012 NRC conducted Phase 2 of the audit at the Westinghouse Cranberry Office in Cranberry Twp, Pennsylvania. This phase of the audit focused on advanced first core (AFC) impacts on shutdown margin (xenon distribution, limiting shutdown margin determination, treatment of enhanced gray rod cluster assemblies, etc.), reactivity coefficients and peaking factor changes, and the follow-up items from phase 1 of the audit.

Pertaining to GSI-191, the staff continued the interviews with Westinghouse personnel and reviews of supporting documentation that began during the first phase of the audit. Westinghouse presented detailed drawings of the egrids and eIFMs, which allowed the staff to understand the differences as compared with the older design. Westinghouse also provided the grid hydraulic test results which supported the findings in WCAP-17524. Discussions were held with Westinghouse personnel to understand the methodology used to determine if a fuel assembly design change was conservative or not from a GSI-191 perspective. The staff concluded that some of the information discussed would be needed on the docket for the staff to reach a final safety conclusion, and RAIs were drafted to request this information.

Regarding the large break LOCA (LBLOCA) calculation, Westinghouse provided the results of the 31-case LBLOCA analysis used to determine the limiting PCT value. Westinghouse also described the methodology used to perform sensitivity analyses with manual F_Q value adjustments. The staff requested additional information regarding the NOTRUMP burnup studies in the SBLOCA analysis to help understand the variations in rod internal pressure relative to burnup as seen in the LBLOCA analysis.

[

]

NRC staff reviewed the rod ejection accident analysis documentation. The methodology is based on a 3D spatial kinetics code package, which is different than what was used for the DCD Revision 15 analysis. The staff generally felt that the documented analysis followed the approved WCAP-15806-P-A methodology, but there was a remaining question regarding the codes allowed by approved methodology. This concern was included in a draft RAI.

Westinghouse personnel provided presentations and discussions to the staff regarding burnup dependent thermal conductivity degradation. Specifically, the staff investigated potential margin impacts for the following fuel design criteria: fuel melt, rod internal pressure, clad stress, steady-state clad strain, transient clad strain, and clad fatigue.

The staff also reviewed the supporting documentation for the non-LOCA accident analyses. The staff noted that various supporting calculation notes contained in what appeared to be open items. The staff notified Westinghouse that all analyses supporting conclusions related to WCAP-17524 must be complete and accurate. Additionally, the staff discussed the results of a locked rotor with LOOP accident analysis. The staff was not provided a detailed analysis of the report. These concerns were included in draft RAIs.

Westinghouse notified the staff that the planned discussion regarding fuel seismic response would be postponed until the 3rd phase of the audit due to scheduling conflicts with the technical staff.

Phase 3 Audit

On September 11-12, 2012, NRC conducted Phase 3 of the audit at the Westinghouse Twinbrook office in Rockville, Maryland. This phase of the audit focused on fuel seismic response and the follow-up items from phase 2.

Westinghouse presented an overview of their approach regarding fuel seismic response. This approach includes the use of credit for reduced impact forces due to hydraulic damping. The staff requested more details including a description of the flow rates to be assumed (and associated damping). Additionally, Westinghouse described the future grid-crush testing plans with relaxed springs which will be used in a final comprehensive fuel seismic supplement. The database used by Westinghouse to develop the dynamic hydraulic damping coefficient was presented to the staff. The staff noted that the data did not include any tests on fuel designs identical to the WCAP-17524 fuel design, and that many of the tests included highly dissimilar fuel designs.

Follow-up discussions were held with Westinghouse regarding the thermal model used in the FIGHT-H code calibration. Westinghouse explained that the TCD model used in FIGHT-H contains the same burnup dependence as the modified PAD 4.0 with TCD. The staff had additional questions regarding gap conductance when compared with PAD. This question was drafted as an RAI.

AP1000 Core Reference Report Request for Additional Information

The following initial set of RAIs were developed from the material covered during all three phases of the audit. Modifications to these questions will still be possible based on future conversations with WEC.

CRR-001 *(spent fuel pool criticality)*

With the movement to the Advanced First Core (AFC), there is a change from a 3-region design to a 5-region design. With respect to spent fuel pool criticality analyses as described in APP-GW-GLR-029, Revision 3, titled, "AP1000 Spent Fuel Storage Racks Criticality Analysis,"

a) How does the change to a 5-region core affect the previously identified limiting fuel assembly depletion characteristics? For example, the current analysis in APP-GW-GLR-029 identifies the limiting assembly insert combination during fuel depletion as having [ ].

b) Considering the AFC and future cycle core designs, how is the limiting assembly insert combination affected?

c) Also, with a change in the core design, it is likely that the axial burnup distributions have also changed. Demonstrate that the change in core design, including the effect of reload core designs, either does not affect the limiting axial burnup distributions as discussed in

APP-GW-GLR-029 or update the safety analyses in APP-GW-GLR-029 to include the appropriate distributions and analysis impacts.

d) Demonstrate that the cumulative impact of the AFC and reload core designs still satisfy the appropriate criteria in 10 CFR 50.68.

CRR-002 *(codes and methods)*

The CRR includes various updates to the referenced codes and methodologies used in the nuclear design. WCAP-10965-P-A, "Qualification of the New Pin Power Recovery Methodology," was added to reflect an updated methodology to be used along with the ANC code. The staff's SER (ML102350046) states that this methodology is acceptable as long as the nuclear data generated as input to ANC originates from the PARAGON and NEXUS code systems. During the Phase 2 audit, staff asked if there were any instances where PHOENIX-P is used instead of PARAGON when generating data for ANC since the SER conclusions for the pin power recovery methodology only give approval for use of the methodology with the PARAGON/NEXUS/ANC code system. Westinghouse stated that data from PHOENIX-P is used in a limited capacity with ANC. For clarity, list all PHOENIX-P uses for calculations supporting updates to FSAR Revision 19 for the CRR and justify why the limitations and conditions in the staff SER mentioned above are not violated.

CRR-003 *(codes and methods)*

The CRR includes an update to DCD Section 15.4.8 describing the rod ejection analysis. The updated DCD section now references the rod ejection calculational methodology described in WCAP-15806-P-A, "Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics," which is based on the standalone SPNOVA code. It appears that the codes supporting the 3-D rod ejection calculation have evolved significantly since staff's 2003 approval of WCAP-15806-P-A. Since the CRR rod ejection analyses are now being performed solely with ANC9.4 due to the migration of the SPNOVA solver into ANC9.4, demonstrate that ANC9.4 produces results that are consistent with previous 3-D rod ejection analyses supported by SPNOVA as a standalone code for the AP1000 design.

CRR-004 *(LBLOCA Analysis)*

In the large-break LOCA analysis (Calculation Note APP-SSAR-GSC-772, Revision 0, "Evaluation of TCD for AP1000 Advanced First Core Application Program and DCD Revision 19 Best-Estimate LBLOCA ASTRUM Analyses,") using the ASTRUM method to evaluate the impact of the fuel thermal conductivity degradation (TCD), the average fuel assembly burnups are limited to [
].

Describe the processes that the average assembly peaking factors and burnups are calculated so as not to underestimate the initial stored energy of the average fuel assemblies. Provide justifications of limiting the burnups of the average assemblies to [ ].

CRR-005 *(LBLOCA Analysis)*

As stated in Supplemental Information to WCAP-17524, "AP1000 Core Reference Report" to Address Thermal Conductivity Degradation" (LTR-NRC-12-46 P-Attachment, June 2012), the large-break LOCA analysis, which accounts for the fuel TCD, is performed with a reduction in

the as-analyzed total peaking factor FQ to a value closer to the desired FQ to remove analysis conservatism. Calculation Note APP-SSAR-GSC-772, Revision 0, states that this input FQ conservatism reduction is done by adjusting sampling values of the power integrals at the bottom and middle 1/3 of the core (i.e., PBOT and PMID) to recover the peak linear heat rate margin in the existing analysis.

(a) Describe the process used for adjusting the PBOT and PMID sampled values to obtain FQ values closer to the desired value.

(b) Explain why this adjusting process is appropriate with respect to the sampling of PBOT/PMID in the ASTRUM methodology.

CRR-006 *(LBLOCA Analysis)*

Supplemental Information to WCAP-17524 (LTR-NRC-12-46 P-Attachment, June 2012) states that the large-break LOCA analysis evaluation of the fuel TCD effects considered peaking factor burndown effects. Table 5-2, "Peaking Factor Assumed in the Evaluation of TCD," of the Supplemental Information provides the values of the peaking factors (FQ and FDH) as a function of rod burnup.

Describe how the values of FQ and FDH in Table 5-2 are determined and how they are implemented in the large-break LOCA analysis.

CRR-007 *(LBLOCA Analysis)*

Calculation Note APP-SSAR-GSC-772 indicated that the large-break LOCA analysis for the evaluation of the fuel TCD effect is performed using RUV reactor coolant pump designed by KSB.

(a) Provide a list of significant differences or changes in the LBLOCA-TCD WCOBRA/TRAC analysis from the analysis in the AP1000 Design Control Document.

(b) Describe why each of these changes is necessary and appropriate.

CRR-008 *(LBLOCA Analysis)*

To properly address fuel thermal conductivity degradation effects on the large-break LOCA analysis for the advanced first core application program, Westinghouse should perform reanalysis of the best-estimate large-break LOCA analysis with total number of runs required by the ASTRUM evaluation method.

CRR-009 *(LBLOCA Analysis)*

It is stated in Supplemental Information to WCAP-17524, "AP1000 Core Reference Report" to address the Impact of Thermal Conductivity Degradation on Additional Events (LTR-NRC-12-56 P-Attachment), that the variations in rod internal pressure relative to burnup are already covered to a large extent in the small-break LOCA burnup studies with NOTRUMP.

Please provide the small-break LOCA burnup studies referred in the Supplemental Information

CRR-010 *(TH design)*

With respect to the AP1000 fuel design that was developed after the NRC approval of WCAP-15063-P-A, Revision 1, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0),"address the following:

a) Discuss the validation including the development of the thermal conductivity degradation (TCD) model used in the interim version of PAD 4.0, called PAD 4.0 TCD, including all coefficients of the TCD equation.

b) There are several versions of the TCD equation. What considerations were used for selecting the TCD equation used in the STAV 7.3 fuel performance code, which is licensed for BWR fuel but not for PWR, for the PAD 4 code?

c) Identify any modifications made to the STAV 7.3 TCD equation to make it compatible to the PAD 4.0 code and PWR fuel designs.

d) Include the plots of thermal conductivity versus temperature at burnups of 0, 20, 40, and 65 GWd/MTU with/without the burnup coefficients default option enabled.

e) Provide a discussion of the predicted PAD 4.0 data/Halden benchmark data set parameters including the initial/test conditions of the data and identify any deviations between the data sets conditions that may impact evaluation results between the data sets.

CRR-011 *(TH design)*

In the PAD 4.0 code, the thermal conductivity correlation is a function of [ ]. The burnup dependent thermal conductivity degradation equation from STAV 7.3 is used in the interim version of PAD 4.0, called PAD 4.0 TCD. The STAV 7.3 code is approved for BWR fuel design analysis and is designed to handle fuel with gadolinium as an integral burnable absorber. The AP1000 fuel design may include axial blankets (fuel pellets of a reduced enrichment), annular fuel pellets in the top and bottom 8 inches of the fuel stack (fully enriched or partially enriched), and integral fuel burnable absorbers (boride-coated fuel pellets or fuel pellets containing gadolinium oxide mixed with uranium oxide). Address the following:

a) What are the impacts that these fuel design features may have on the thermal conductivity correlation?

b) Provide plots of the PAD 4 TCD predicted temperature profile data and the Halden benchmark data with/without the fuel burnup coefficient enabled.

c) Discuss any differences between the data profiles.

d) Discuss the applicability of the burnup dependent thermal conductivity degradation equation from STAV 7.3 for IFBA coated pellets.

CRR-012 NOT USED

CRR-013 *(GSI-191)*

WCAP-17524-P provides a description of fuel design changes to be included in the advanced first core for the AP1000 plant design. Section 2.6 discusses the impacts of the protective grid design change on GSI-191 but does not discuss the impacts of other fuel design changes. Provide an analysis of the impacts on GSI-191 for each design change, including the eIFM and eMidGrids.

CRR-014 *(GSI-191)*

In order to support the staff's review of the new fuel assembly design impacts on GSI-191, provide a comparison of the fuel designs for the following:

a) AP1000 DCD Revision 19

b) Test assembly used to support AP1000 DCD Revision 19

c) Fuel assembly design for WCAP-17524-P

d) Test assembly for PWROG which included the Robust P-Grid.

Also provide the test conditions for both the AP1000 DCD Revision 19 tests and the PWROG tests used to support the Robust P-Grid (e.g. glow rates, debris types/quantities, etc.).

CRR-015 *(GSI-191)*

What is the screening methodology used by Westinghouse to determine the impacts on GSI-191 for each fuel design change.

CRR-016 *(Calc Notes-Open Items)*

During the audit held Aug 14-16, 2012, it was noted that some of the calc notes used to support WCAP-17524-P contain open items. The submitted topical report must be based on final analyses following approved quality assurance programs. Therefore, close all related open items and update WCAP-17524-P if necessary.

CRR-017 *(thermal conductivity degradation)*

The PAD 4.0 code incorrectly excludes burn-up effects in its thermal conductivity model. By using a corrected thermal conductivity model and the existing fission gas model, the code could overestimate the amount of fission gas and the fuel rod internal pressure since the current fission gas model was correlated to the testing data using the original thermal conductivity model. To correct this, [

], it is unknown if the calculated amount of fission gas released remain conservative for AP1000 applications, considering the differences in geometry, burn-up levels, and power history between the test assemblies used to calculate the fission gas release model and the AP1000 Core Reference

Report application design. Demonstrate that this assumption will lead to conservative or accurate predictions of fission gas quantities and fuel rod internal pressure within the power and burn-up range of this application.

CRR-018 *(non-LOCA accidents)*

As part of this topical report, Westinghouse developed a new procedure to mitigate the vessel head vent opening event by discharging high temperature and high pressure primary coolant into the in-containment RWST. Similar approaches have been developed for loss of feedwater heater event, inadvertent opening of CMT valves and CVCS malfunction. It was indicated by Westinghouse that the discharge of primary coolant into the containment will cause high containment air space temperature. Demonstrate that the in-containment equipment qualification program has already taken into account these containment heat up events and that the maximum allowable containment air space temperature for all the equipments required for operation is higher than the maximum containment air space temperature during these AOO events.

CRR-019 *(non-LOCA accidents)*

The rod ejection accident was analyzed using the previous fuel pellet thermal conductivity model, which does not account for burnup effects. With the new pellet thermal conductivity model, it is expected that the fuel pellet initial steady state temperature is significantly higher than what was calculated before, especially for the end-of-cycle condition. What is the quantitative impact of the corrected TCD model with burnup dependence on the rod ejection accident analysis? In particular, provide the impact on the most limiting DNBR, fuel center line temperature and cladding strain. Demonstrate that they all satisfy the relevant limits.

CRR-020 *(non-LOCA accidents)*

A 3-D kinetics code is used to analyze the AP1000 rod ejection accident. Will the same code be used for the future reload analysis?

CRR-021 *(non-LOCA accidents)*

The FIGHT-H code has been used to support transient analysis to provide fuel rod temperature data at different power and burn-up levels. The accuracy of the code affects the calculated Doppler Feedback Effect and the power level for AOOs and rod ejection accident. Evaluate the difference between the FIGHT-H code and the fuel performance code which models the thermal conductivity degradation properly. Based on the evaluation, determine the limitations of the FIGHT-H code and the impact on the calculated DNBR, peak linear power density, transient power level and cladding strain.

CRR-022 *(non-LOCA accidents)*

Loss of flow and LOOP events are the most limiting in terms of system pressure with a margin of about 50 psid before the thermal conductivity degradation is considered. Westinghouse did not anticipate that the peak system pressure is going to exceed the limit if the thermal conductivity degradation is properly modeled; however, the technical basis supporting the conclusion has not been presented. Provide quantitative evidence to justify this conclusion.

CRR-023 *(LOCA)*

Westinghouse has used the previously approved containment mass and energy methodology to perform containment peak pressure calculation during LOCA. Does the corrected thermal conductivity degradation model affect the peak containment pressure and temperature calculation?

CRR-024 (Fuel Seismic)

What is the basis for assuming the applicability of the Westinghouse damping test data from older fuel assembly designs to the AP1000 Core Reference Report fuel assembly design?

CRR-025 (Fuel Seismic)

Provide justification for the applicability of crush test data/analysis from WCAP-12488-P-A Addendum 1, Revision 1 to the eMidGrids and eIFMs used in the AP1000 Core Reference Report fuel assembly design.

CRR-026 *(non-LOCA accidents)*

An accurate or a conservative prediction of core initial stored energy is important to the containment peak pressure calculation for a postulated large break loss-of-coolant accident. The core initial stored energy used as input to this calculation is claimed to be the highest at [ ] in contrast to [ ]. Describe the core initial stored energy calculation method and explain how the core at [ ] rated power can be the most limiting in regards to initial core stored energy.

CRR-027 (Fuel Seismic)

What is the hydraulic damping coefficient that Westinghouse plans to credit for different flow conditions in the fuel assembly seismic response analysis? How would a coast down be handled?