June 14, 2013

Mr. William Bedont, Resource Manager
Westinghouse Electric Company Nuclear Services
P.O. Box 158 – Waltz Mill Site
Madison, PA 15663

SUBJECT: NUCLEAR REGULATORY COMMISSION INSPECTION REPORT NO. 99900404/2013-203, NOTICE OF NONCONFORMANCE

Dear Mr. Bedont:

On April 3-5, 2013, the U.S. Nuclear Regulatory Commission (NRC) staff conducted an inspection at the Westinghouse Electric Company (Westinghouse) facility in Madison, PA. The purpose of the limited scope reactive inspection was to assess Westinghouse's compliance with selected portions of the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) Appendix B, "Quality Assurance Program Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." The inspection focused on design control activities related to the reactor internals flow-induced vibration analysis for the AP1000 pressurized water reactor design. The enclosed report presents the results of this inspection. This NRC inspection report does not constitute NRC endorsement of your overall quality assurance program.

The NRC inspection team sampled design control activities and concluded that Westinghouse is generally effective in implementing its 10 CFR 50 Appendix B program in support of the AP1000 reactor internals flow-induced vibration analysis. However, the NRC inspectors found that the implementation of your quality assurance program failed to meet certain NRC requirements imposed on you by your customers or NRC licensees. Specifically, Westinghouse failed to sufficiently verify or check the adequacy of the flow-induced vibration analyses to assure that the reactor internals are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. In addition, Westinghouse failed to correct a condition adverse to quality related to not applying the random turbulence loads to the vortex suppression plate when Westinghouse used engineering judgment that did not support the basis for this decision. These nonconformances are cited in the enclosed Notice of Nonconformance (NON) and the circumstances surrounding them are described in detail in the enclosed inspection report.

Please provide a written explanation or statement within 30 days of this letter in accordance with the instructions specified in the enclosed NON. We will consider extending the response time if you show good cause for us to do so.

In accordance with 10 CFR 2.390 "Public Inspections, Exemptions, Requests for Withholding," of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's Agencywide Documents Access and Management System, accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request that such material is withheld from public disclosure, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21 "Protection of Safeguards Information: Performance Requirements."

Sincerely,

/RA/

Edward H. Roach, Chief
Mechanical Vendor Branch
Division of Construction Inspection
and Operational Programs
Office of New Reactors

Docket No. 99900404

Enclosures:
1. Notice of Nonconformance
2. Inspection Report No. 99900404/2013-203
and Attachment

the NRC's Agencywide Documents Access and Management System, accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request that such material is withheld from public disclosure, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21 "Protection of Safeguards Information: Performance Requirements."

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Docket No. 99900404

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1. Notice of Nonconformance
2. Inspection Report No. 99900404/2013-203
and Attachment

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NOTICE OF NONCONFORMANCE

Westinghouse Electric Company Docket No.: 99900404
Madison, PA. Inspection Report No.: 99900404/2013-203

Based on the results of a Nuclear Regulatory Commission (NRC) inspection conducted at the Westinghouse Waltz Mill (Westinghouse) facility in Madison, PA, on April 3 -5, 2013, certain activities were not conducted in accordance with NRC requirements, which were contractually imposed on Westinghouse by its customers:

A. Criterion III, "Design Control," of Appendix B, "Quality Assurance Program Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR 50 states, in part, that the design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.

   Contrary to the above, as of April 5, 2013, Westinghouse failed to sufficiently verify or check the adequacy of the flow-induced vibration analyses to assure that the reactor internals are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. Specifically, Westinghouse failed to adequately address the 15% difference between the core barrel first beam mode frequencies in the reactor coolant pump (RCP) reactor equipment system model (RESM) and the random turbulence RESM, which is greater than the 10% industry accepted standard. The RCP RESM and the random turbulence RESM results are used to predict the high cycle fatigue stress of reactor vessel internal components and to ensure that the high cycle fatigue stress endurance limits in the 1998 Edition of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Mandatory Appendix I, Figure I-9.2.2, "Design Fatigue Curves for Austenitic Steels, Nickel-Chromium-Iron Alloy, Nickel-Iron-Chromium Alloy, and Nickel-Copper Alloy for Sa ≤ 28.2 ksi, for Temperatures Not Exceeding 800°F" are met. In addition, the 10% and 3.5% frequency sweeps applied around the RCP forcing frequencies in the RCP pulsation analysis to account for uncertainties in pump speed, fluid density, and sound speed do not cover the 15% uncertainty core barrel first beam mode frequencies in the two RESMs.

This issue has been identified as Nonconformance 99900404/2013-203-01.

B. Criterion XVI, "Corrective Action," of Appendix B, to 10 CFR 50 states, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected.

   Contrary to the above, as of April 5, 2013, Westinghouse failed to correct a condition adverse to quality. Specifically, Westinghouse used incorrect engineering judgment to close Corrective Action Process (CAPs) Issue Report (IR) 12-286-W001, which addressed not applying random turbulence loads to the base plate in the random turbulence RESM. Westinghouse closed CAPs IR 12-286-W001 and revised the calculation note in APP-MI01-S3C-331, Rev. 3, "Flow-Induced Vibration (FIV) of the AP1000 Vortex Suppression Plate and Secondary Core Support Structures," based on engineering judgment that assumed additional conservatism in the analysis that did not exist. Westinghouse asserted that applying the random turbulence pressure load in the upwards and downwards direction to the vortex suppression plate would effectively double the applied load, which it did not.

This issue has been identified as Nonconformance 99900404/2013-203-02.

Please provide a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Chief, Mechanical Vendor Branch, Division of Construction Inspection and Operational Programs, Office of New Reactors, within 30 days of the date of the letter transmitting this Notice of Nonconformance. This reply should be clearly marked as a "Reply to a Notice of Nonconformance" and should include for each noncompliance: (1) the reason for the noncompliance, or if contested, the basis for disputing the noncompliance; (2) the corrective steps that have been taken and the results achieved; (3) the corrective steps that will be taken to avoid noncompliance; and (4) the date when your corrective action will be completed. Where good cause is shown, the NRC will consider extending the response time.

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's Agencywide Documents Access and Management System, which is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html, to the extent possible, it should not include any personal privacy, proprietary, or Safeguards Information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request that such material be withheld, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If Safeguards Information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21, "Protection of Safeguards Information: Performance Requirements."

Dated this the 14th day of June 2013.

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NEW REACTORS
DIVISION OF CONSTRUCTION INSPECTION AND OPERATIONAL PROGRAMS
VENDOR INSPECTION REPORT

Docket No.:	99900404
Report No.:	99900404/2013-203
Vendor:	Westinghouse Electric Company Nuclear Services P.O. Box 158 – Waltz Mill Site Madison, PA 15663
Vendor Contact:	Mr. William Bedont Process Manager Telephone: (724) 722-6731 E-mailbedontwj@westinghouse.com
Nuclear Industry Activity:	Westinghouse Electric Company (Westinghouse) holds a design certificate for the AP1000 and is responsible for detailed design and testing for safety-related components to be used in AP1000 plants. The detailed design includes reactor vessel internals vibration tests, calculations, and analyses.
Inspection Dates:	April 3 -5, 2013
Inspectors:	Richard McIntyre NRO/DCIP/CMVB, Team Leader Samantha Crane NRO/DCIP/CMVB Yuken Wong NRO/DC/EMB, Technical Specialist
Approved by:	Edward H. Roach, Chief Mechanical Vendor Branch Division of Construction Inspection and Operational Programs Office of New Reactors

EXECUTIVE SUMMARY

Westinghouse Electric Company
99900404/2013-203

The U.S. Nuclear Regulatory Commission (NRC) inspection focused on the implementation of the Westinghouse Electric Company (Westinghouse) design control process as it pertains to addressing the effects of design changes on the AP1000 reactor internals flow induced vibration (FIV) analyses. The purpose of this inspection was to verify that Westinghouse implemented an adequate quality assurance (QA) program in support of FIV analyses that complied with the requirements of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities." The NRC inspectors conducted the inspection at the Westinghouse facility in Madison, PA, on April 3-5, 2013.

During this inspection, the NRC inspection team followed-up on specific issues relating to the technical adequacy of the reactor internals FIV analyses that were identified in several corrective action process (CAP) issue reports (IR): CAPs IR 12-286-W003, 13-045-M035, 12-285-M038, 12-286-W001, 13-029-M049, and 13-072-M014. These CAPs document the Westinghouse engineering judgment used to justify not including certain design changes in the reactor internals FIV analyses, as well as specific issues related to the adequacy of the reactor internals FIV analyses The NRC inspection team interviewed Westinghouse technical staff and reviewed associated calculations, design reports, analyses, and test reports to verify that all design changes to the reactor internals were sufficiently modeled or evaluated, and the FIV loads are adequately addressed in the analyses. Prior to the inspection, the NRC inspection team electronically submitted a number of questions related to the AP1000 comprehensive vibration assessment program (CVAP) to be used to focus the inspection. The questions are included in attachment to this inspection report.

The following regulations served as the bases for the NRC inspection:

- Appendix B of 10 CFR Part 50

During the conduct of this inspection, the NRC inspectors implemented Inspection Procedure (IP) 43003, "Reactive Inspections of Nuclear Vendors."

The NRC inspectors determined that Westinghouse's responses adequately addressed the 17 questions identified in the attachment to this inspection report. With the exception of Nonconformance 99900404/2013-203-01, Westinghouse was able to quantitatively demonstrate the soundness of its engineering judgment used to justify not including certain design changes and loads in the flow induced vibration analysis. For licensees that reference the approved AP1000 design, the NRC staff will have the opportunity to evaluate the AP1000 CVAP predictive analysis prior to preoperational reactor internals flow-induced vibration testing.

Nonconformance 99900404/2013-203-01 was issued for Westinghouse's failure to verify the adequacy of the following models and tests used to predict the AP1000 reactor internals dynamic response: the reactor coolant pump reactor equipment system model (RESM) and the random turbulence RESM. Nonconformance 99900404/2013-203-02 was issued for Westinghouse's failure to correct a condition adverse to quality related to not applying the random turbulence loads to the base plate when Westinghouse used engineering judgment that did not support the basis for this decision.

REPORT DETAILS

1. Design Control

a. Inspection Scope

The inspectors reviewed the implementation of the Westinghouse design control process as it pertains to addressing the effects of reactor internals design changes on flow-induced vibration (FIV) analyses. The inspectors interviewed Westinghouse personnel and reviewed relevant documentation to ensure that Westinghouse addressed the 17 questions identified in the attachment to this inspection report. The documents reviewed by the inspectors are also included in the attachment to this inspection report.

b. Observations and Findings

b.1 Design Changes in the Reactor Equipment System Models (RESM) for the Reactor Coolant Pump (RCP) Pulsation Analysis and Random Turbulence Analysis

The NRC inspectors interviewed Westinghouse staff to determine the major design changes made to the AP1000 reactor internals since 2003. The changes include:

- Change in the core shroud design details
- Increase in reactor vessel inner diameter below the nozzles
- Addition of reactor vessel flow skirt
- Incorporation of instrumentation grid assembly (IGA)
- Relocation of radial support keys from 45° locations to the cardinal axes

The NRC inspectors interviewed the Westinghouse staff to determine if the random turbulence Reactor Equipment System Models (RESM) includes the design changes describe above. The NRC inspectors reviewed the RESM for the Reactor Coolant Pump (RCP) pulsation analysis presented in Revision 2 of APP-RXS-M3C-029, "AP1000 RVI RCP-Induced Vibration," and CAPs IR 12-285-M038, to verify that the design changes were either incorporated into the RCP RESM or that evaluations and analyses were performed that provided adequate justification for not including the design changes in the RCP RESM.

To address the NRC inspectors' concerns identified during the inspection with regards to the engineering justification for not including all of the design changes in the RCP RESM, Westinghouse performed a sensitivity study to assess the analyses results that included the reactor vessel inside diameter increase, incorporation of the IGA, and the addition of flow skirt design changes to the RCP RESM.

The sensitivity study indicates that with the design changes incorporated, the core barrel first beam mode frequency increased 2%. In the modeling of the hot functional test using the100% pump speed condition, the core barrel flange RCP pulsation-induced shear load increased 7%, while the combined random turbulence and RCP pulsation shear loads increased by 2%. The secondary core support structures (SCSS) analysis, APP-MI01-S3C-331, Rev. 3, "Flow-Induced Vibration (FIV) of the AP1000 Vortex Suppression Plate and Secondary Core Support Structures," is based on the response of the lower core support plate (LCSP). As a result of incorporating the design changes in the analysis, the displacement of the LCSP at the second pump harmonics increased approximately 13% in the horizontal direction and 5% in the vertical direction. In determining the impact of this increase, all pump frequencies were compared to the natural frequencies of the SCSS for the first three modes, and found that they do not coincide when the 10 percent frequency sweep is included. To further demonstrate that the increases are insignificant, an explicit analysis was performed and documented in Revision 0 of CN-ARIDA-13-5, "AP1000 RVI RCP-Induced Vibration with Updated Design Changes" to show that the force and moment reactions increased a maximum of 3 percent and 2 percent, respectively. Therefore, Westinghouse concluded that including the design changes does not significantly affect the results.

The NRC inspectors determined that the sensitivity study provides quantitative results to demonstrate with reasonable assurance that the design changes will not cause the reactor internal components to exceed the Mandatory Appendix I fatigue limits in Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code). Additionally, in other assessments discussed later in this report, Westinghouse showed that fatigue stress margin increased when actual FIV loads were applied. No findings of significance were identified.

b.2 <u>The first core barrel beam mode frequency difference between the RCP RESM and the Random Turbulence RESM</u>

The NRC inspectors reviewed the RCP RESM results described in APP-RXS-M3C-029, Rev. 2, "AP1000 RVI RCP-Induced Vibration", the random turbulence RESM results presented in APP-MI01-S3C-107, Rev. 1, "AP1000 RESM – Flow-Induced Vibration Response Analysis," and the 1/7th scale test documented in APFIV-DRT-B21-MV006, Rev. 0, "AP1000 Scale Model Testing FIV Test Report," to verify that the RCP RESM and random turbulence RESM were accurate in predicting the AP1000 reactor internals dynamic response. The core barrel first beam mode frequency in the RCP RESM of 8.4 Hz is 15% higher than the frequency in the random turbulence RESM of 7.3 Hz. In addition, the core barrel first beam mode frequency in the RCP RESM is 11% higher than the frequency in the 1/7 scale test of 7.57 Hz. Westinghouse stated that the frequency of the core barrel first beam mode is well below the first discrete pump frequency (e.g., first pump harmonic frequency at 100% flow is approximately 30 Hz) and therefore the difference in the core barrel first beam mode in the RCP RESM and the random turbulence RESM will have a negligible impact on the core barrel response in the RCP loadings.

The NRC inspectors noted that the acoustic forcing function development in APP-RXS-M3C-303, Rev. 0, "Calculation of Pump-Induced Pulsation Loads for AP1000 Reactor Vessel Internals," includes a frequency sweep around the RCP forcing frequencies to account for uncertainties in pump speed, fluid density, and sound speed. For isothermal reactor coolant system conditions such as hot standby, the frequency sweep is +/-3.5%. The frequency sweep at the normal operating conditions is +/-10%. Westinghouse stated that the basis for using the frequency sweep of +/-3.5% for the isothermal condition is less uncertainty in the pump speed, sound speed, and fluid property.

The NRC inspectors identified that the core barrel first beam mode frequency differs significantly between the RCP RESM and the random turbulence RESM. In addition, the

NRC inspectors identified that the 3.5% frequency sweep for isothermal RCS conditions is less than the accepted industry standard 10% frequency sweep.

To address the NRC inspectors' concerns, Westinghouse performed a sensitivity study to assess the impact of the frequency variation on the analysis results. The sensitivity study included the design changes discussed in section 1.b.1 above. In the sensitivity study, the core first beam mode frequency in the RCP RESM increased by 2%, thus adding to the frequency difference between RCP RESM and random turbulence RESM frequency predictions.

The greater than 15% difference in the core barrel first beam mode for the two RESMs, which is more than the industry accepted standard of 10%, indicates that RESMs may contain modeling errors. In addition, the 10% and 3.5% frequency sweeps applied in the RCP pulsation analysis do not cover the 15% uncertainty in the two RESMs. The NRC inspectors identified this issue as Nonconformance 99900404/2013-203-01 for failure to verify or check the adequacy of the design. For licensees that reference the approved AP1000 design, the NRC will have the opportunity to evaluate the CVAP predictive analysis prior to preoperational reactor internals flow-induced vibration testing.

b.3 <u>Core Barrel Displacement at Lower Core Support Plate (LCSP)</u>

The NRC inspectors reviewed CAPs IR 12-286-W003 to verify that Westinghouse appropriately addressed the issue regarding core barrel displacement at the LCSP being smaller than the displacement at the core barrel mid-span elevation in the random turbulence RESM. CAPs IR 12-286-W003 had been previously closed based on the conclusion that the model was behaving as expected using the forcing functions that were applied, and that similar behavior was observed when the core barrel response at the LCSP in the RESM was compared to the response in the 1/7 scale test.

In February 2013, after CAPs IR 12-286-W003 was closed, Westinghouse discovered an error in the methods used to apply the forcing functions to the core barrel, reactor vessel, and reactor vessel lower head in Revision 1 of APP-MI01-S3C-107. All turbulence power spectral density (PSD) loads on these components were inadvertently applied in-phase. Applying the loads in-phase enhanced the shell mode response of the core barrel, while suppressing the beam mode response. Thus, the core barrel shell mode response and the beam mode response were inaccurate in a non-conservative manner.

Additionally, Westinghouse discovered data errors in the Toshiba 1/7th scale test report, APFIV-DRT-B21-MV006, Revision 0, "AP1000 Scale Model Testing FIV Test Report," caused by faulty sensor charge amplifiers. Westinghouse stated that the raw accelerometer data from the Toshiba 1/7th scale test report are integrated to obtain the core barrel displacement. The Toshiba 1/7th scale test data error led to the conclusion that the core barrel displacement at the LCSP is smaller than the core barrel wall displacement at mid-span elevation. The test results, including the core barrel displacements, are documented in the Westinghouse test report for the 1/7th scale test.

Westinghouse took immediate corrective action and opened CAPs IR 13-045-M035 to address the forcing function application in the random turbulence RESM. Preliminary results with corrected turbulence loadings indicate that the core barrel beam mode is dominant and the core barrel response is now consistent with past test data. Review of these results and the potential impact on design and CVAP analyses are ongoing.

b.4 RCP Pulsation Loads on the SCSS Columns, Energy Absorbers and Base Plate

The NRC inspectors reviewed APP-MI01-S3C-331, Rev. 3, “Flow-Induced Vibration (FIV) of the AP1000 Vortex Suppression Plate and Secondary Core Support Structures,” that documents the RCP pulsation loads on the SCSS, including the energy absorbers and base plate, to verify that the RCP pulsation loadings are insignificant for the SCSS compared to the total FIV loadings, which are dominated by random turbulence. The analysis considers RCP pulsation loadings applied to the vortex suppression plate (VSP) only. The sensitivity analyses in APP-MI01-S3C-331 show that the increase in reaction loads would be negligible if RCP loadings were applied to the SCSS columns in addition to the VSP. Westinghouse stated that applying RCP pulsation loadings to the energy absorbers and base plate is expected to result in a negligible increase in reactions. This assertion is based on the results of the sensitivity analysis in APP-MI01-S3C-331 with the loadings applied to the columns, as well as engineering judgment that the acoustic pressures acting on the energy absorbers and base plate would generally be lower than those acting on the VSP or columns. A subsequent sensitivity study that includes all the actual RCP pulsation loads and random turbulence loads determined that the results do not change significantly. The sensitivity study for the SCSS component is further discussed in Section 1.b.5 of this report.

To address the NRC inspectors’ concerns identified during the inspection with respect to whether the natural frequencies of the SCSS are outside the RCP forcing frequencies for the bounding operating conditions, Westinghouse performed a sensitivity study that included all design changes. The sensitivity study compared the natural frequencies of the SCSS to the RCP forcing frequencies for the bounding conditions. In the sensitivity study, the RCP RESM predicted that the SCSS natural frequencies for the first 3 modes are outside of all RCP frequencies, including the 10 percent frequency sweep. No issues of significance were identified.

b.5 Random Turbulence Loads on VSP and Base Plate

The NRC inspectors reviewed CAPs IR 12-286-W001 to verify that the justification for not including the random turbulence loads applied to the base plate was appropriately dispositioned. The CAPs IR resolution states: “A discussion was incorporated into Section H.2 of APP-MI01-S3C-331 Rev. 3 that states the random turbulence pressure load was applied in the upwards and downwards directions to the VSP, which effectively doubles the load applied to the VSP. Given that the area of the base plate is much smaller than the area of the VSP, applying the pressure PSD to the VSP and base plate in a single direction, rather than to the VSP in both the upward and downward directions would result in an overall lower force applied to the SCSS.”

Westinghouse clarified that alternating PSD random turbulence loadings acting on both the top and bottom surfaces of the VSP are considered in the current SCSS analysis in APP-MI01-S3C-331, Rev. 3. While the bottom surface is expected to experience lower loadings due to lower velocities in the region of the lower head, the random turbulence PSD loadings in the analysis are conservatively applied with the same magnitude to both surfaces of the VSP. The VSP is analyzed with PSD load cases applied separately in the upward and downward directions. Each load case imposes fully-reversing loads on the structure. Thus, the upward and downward load cases are equivalent to fully-reversing loads on the top and bottom surfaces of the VSP. Because these load cases are due to random turbulence and are statistically uncorrelated, the results are combined by the square root of the sum of the squares.

The NRC inspectors identified that turbulence forces are dynamic forces that act in both directions (i.e., upwards and downwards) and therefore the loads applied to the VSP would not be doubled. The NRC staff determined that the engineering judgment did not support the basis for closing CAPs IR 12-286-W001 and the basis for not applying the random turbulence loads to the base plate in APP-MI01-S3C-331, Rev. 3. In further discussions with Westinghouse staff, they recognized that the loads applied to the VSP are not doubled. However, Westinghouse was able to provide assurance that while the engineering justification provided in CAPs IR 12-286-W001 and the calculation note in APP-MI01-S3C-331, Rev. 3 are incorrect, the loadings applied in the analysis were conservative. The NRC inspectors identified this issue as Nonconformance 99900404/2013-203-02 for failure to correct a condition adverse to quality.

The NRC inspectors also reviewed the Westinghouse engineering judgment used to justify not applying vertical random turbulence PSDs to the base plate in the SCSS analysis in APP-MI01-S3C-331, Rev. 3. The justification is based on the low velocities expected in the bottom of the lower head. Additionally, the analysis includes a conservative random turbulence loading applied vertically to the VSP. Westinghouse concluded that additional loading on the base plate would be offset by the conservatisms already included in the analysis in the vertical VSP loadings. Similar conservatisms in the flow skirt turbulence pressures include neglecting any velocity decay from the exit of the flow skirt to the SCSS. Since the turbulence scales approximately as the square of the velocity, Westinghouse concluded that the current VSP vertical loadings are significantly conservative.

The NRC inspectors noted that velocity is only one of the parameters in determining the random turbulence response. The frequency of the random turbulence forcing function is also important in determining whether resonance occurs when the frequency of the forcing function coincides with one of the natural frequencies of the structures.

To address the NRC inspectors' concerns with respect to the random turbulence loads on VSP and base plate, Westinghouse performed a sensitivity study to support the engineering judgment presented in the random turbulence analysis of the SCSS. The sensitivity study includes actual loadings based on the reactor coolant velocities at the exit of the flow skirt, and its propagation through the lower plenum. The sensitivity study applies the actual RCP and random turbulence loads to all surfaces of the SCSS components (support columns, VSP, energy absorbers, and base plate). Per APP-MI01-S3R-002, Rev. 3, "AP1000 Generic Reactor Vessel Internals (RVI) Summary Design Report," the energy absorber weld has the lowest high cycle fatigue stress margin among the SCSS components. The sensitivity study presents the reaction loads at the energy absorber as reaction loads at the base plate because the stress in the energy absorber welds are directly proportional to the reaction loads applied to the energy absorbers. The results of the sensitivity study determined that the reaction loads using actual loads are at least 36% less than the reaction loads that used conservative design loads, as presented in APP-MI01-S3C-331, Rev. 3. In conclusion, the sensitivity study quantitatively demonstrates that the existing analysis results are conservative and that the SCSS component fatigue stresses will not exceed the ASME Code stress limits when the appropriate RCP and random turbulence loads are applied to the SCSS components.

b.6 Random Turbulence load on the Core Shroud

The NRC inspectors reviewed the white paper in the attachment to APP-MI01-S3C-107, Rev. 1 to verify that the engineering judgment used to justify not applying the random turbulence load to the core shroud was adequate. The white paper concludes that the forcing function on the core shroud is only 1% of the forcing function on the core barrel. In addition, Westinghouse stated that the direct loadings due to random turbulence forces in the reactor core region are small compared to the loads induced by the hydraulic coupling between the core shroud and core barrel. The direct loadings due to random turbulence forces in the reactor core region are also small compared to the base excitation driven by the vibration of the LCSP due to random turbulence. Lastly, Westinghouse stated that the forcing function on the core shroud is negligible compare to the forcing function on the core barrel.

To provide additional quantitative basis to support the engineering judgment to justify not including the random turbulence load on the core shroud, Westinghouse provided a comparison of the analytical results from the random turbulence RESM and 1/7th scale flow test results. The comparison shows that both the core shroud-to-core barrel root mean square (RMS) displacement and the absolute LCSP RMS displacement from the random turbulence RESM are at least 1.44 times greater than those in the AP1000 1/7th scale test. Westinghouse stated that this comparison demonstrates that even with the exclusion of the core shroud turbulence forcing functions from the random turbulence RESM, the response of the LCSP is still conservative.

Westinghouse also performed a sensitivity study that repeated the random turbulence RESM analysis with the inclusion of the direct application of core-side turbulence loads to the core shroud. The relative core shroud-to-core barrel displacement and LCSP lateral displacements from the sensitivity study were compared to the results from Revision 2 of APP-MI01-S3C-107 that does not include the direct-applied core shroud loads. The comparison shows that the displacements from the new analysis increase by no more than 0.1%. Westinghouse concluded that the direct-applied turbulence loads on the core shroud has a negligible effect on the response of the core shroud and the LCSP. The quantitative results from the sensitivity study demonstrated that the direct-applied random turbulence loads on the core shroud has a negligible effect on the FIV analysis results. No issues of significance were identified.

b.7 Use of ASME Code Figure I-9.2.2 Fatigue Curves

Revision 19 of the AP1000 certified design uses the 1998 Edition through 2000 Addenda of the ASME Code as the ASME Code of record. The NRC inspectors reviewed APP-MI01-S3R-002, Rev. 3, to verify that Westinghouse used the appropriate fatigue curves in the 1998 Edition of the ASME Code, Section III, Article NG-3000, "Design," Appendix I, Figure I-9.2.2, "Design Fatigue Curves for Austenitic Steels, Nickel-Chromium-Iron Alloy, Nickel-Iron-Chromium Alloy, and Nickel-Copper Alloy for $S_a \leq 28.2$ ksi, for Temperatures Not Exceeding 800°F." Mandatory Appendix I allows the use of three separate fatigue curves if the requirements in the Code are met. The three curves included the less conservative Curves A, B and the more conservative Curve C. The less conservative Curves A and B were removed from the Code starting in the 2007 Edition.

The RVI summary design report uses the Curve A endurance limit for the high cycle fatigue stresses for the guide tube support pins and direct vessel injection deflector support pins. All other stainless components are below the Curve C endurance limit. Using Curve A for the pins meets the Code requirements because the pins are machined and are non-welded components. No issues of significance were identified.

b.8 Preoperational Tests for Upper Internals Components.

The AP1000 CVAP analysis and test planning are currently in progress. The NRC inspectors interviewed Westinghouse staff to determine how the Westinghouse test plan will meet Regulatory Guide (RG) 1.20, Rev. 2, “Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing.”

Westinghouse described the flow paths to the upper reactor vessel head. The flow through the spray nozzles is driven by the hydraulic pressure differential between the top of the downcomer and the upper head. The fuel pressure drop represents a significant part of the spray nozzle driving pressure differential. Preoperational testing is performed without any fuel loaded in the core. During preoperational testing, the hydraulic pressure loss of the fuel will be simulated by a system of orifice plates installed in the upper internals and upper head flow path. These orifice plates are designed to produce a best estimate of the full RCS flow rate at 100% of the RCP pump speed, while providing prototypic spray nozzle driving pressure drops and velocities. The use of the orifice plates is designed to produce the maximum flow rates possible throughout all areas of the RVI for the 100% RCP pump speed condition.

Since the AP1000 CVAP analysis and test planning are currently in progress, Westinghouse stated that if any planned test conditions are identified for which normal operating conditions are not accurately or sufficiently simulated, these conditions will be explicitly identified and documented. Furthermore, Westinghouse stated that if non-conservatisms are identified, both analytical and testing methodologies to account for those non-conservatisms will be detailed and documented within the CVAP. No issues of significance were identified.

b.9 Computational Fluid Dynamics (CFD) Benchmarking

The NRC inspectors interviewed the Westinghouse staff to verify that the computational fluid dynamics analysis was appropriately benchmarked. Benchmarking is addressed generically in the Westinghouse procedures for design analyses and design verification. For a sample of forcing function calculations, including the CFD analyses, the NRC inspectors verified that these calculations were performed in accordance with relevant Westinghouse procedures. Westinghouse provided an example roadmap of the development of a benchmarked AP1000 downcomer forcing function in which CFD analyses were involved.

The initial AP1000 downcomer turbulence forcing function was developed using a CFD model of the inlet nozzles, downcomer and lower plenum combined with test data from a model scale test of the RVIs for a reference plant that is very similar to the AP1000 RVI configuration. The resulting downcomer turbulence forcing functions were applied to a detailed finite element model (FEM) of the full scale reactor internals of the reference plant. The resulting FEM forced response of the core barrel was compared to the preoperational test core barrel response of the reference plant and the turbulence forcing function was adjusted to make the FEM forced response correlate with the test data.

An updated downcomer turbulence forcing functions was then developed to incorporate even more AP1000 specific test data for comparison, and because of geometry and hydraulic differences between the AP1000 and reference plant. The development of the updated AP1000 downcomer turbulence forcing function used an updated AP1000 CFD model as well as AP1000 specific oscillatory pressure data. The updated AP1000 downcomer turbulence forcing function was then compared to the benchmarked legacy downcomer turbulence forcing function and the enveloping values are used as the final design inputs.

The NRC inspectors verified that the AP1000 turbulence forcing function approach is based on published methodology (M. K. Au-Yang, "Flow Induced Vibration of Power and Process Plant Components," ASME Press, New York, 2001), much of which is derived from model scale test data. The pressure values in this textbook are very conservative to account for the high variability in the reported test data. To provide a more appropriate pressure spectra estimate, the AP1000 turbulence forcing function employs spatially appropriate pressure spectra derived from Westinghouse model scale test data and CFD analyses. No issues of significance were identified.

c. Conclusions

The NRC inspectors concluded that Westinghouse's responses adequately addressed the 17 questions identified in the attachment to this inspection report. With the exception of Nonconformance 99900404/2013-203-01, Westinghouse was able to quantitatively demonstrate the soundness of its engineering judgment used to justify not including certain design changes and loads in the FIV analysis. Nonconformance 99900404/2013-203-01 was issued for Westinghouse's failure to verify the adequacy of the following models used to predict the AP1000 reactor internals dynamic response: the reactor coolant pump RESM and the random turbulence RESM.

Additionally, Nonconformance 99900404/2013-203-02 was issued for Westinghouse's failure to correct a condition adverse to quality related to not applying the random turbulence loads to the vortex suppression plate when Westinghouse used engineering judgment that did not support the basis for this decision.

2. Training and Qualification of Personnel

a. Inspection Scope

The inspectors reviewed Westinghouse's policies and procedures to verify that Westinghouse was implementing training activities in a manner consistent with regulatory requirements and industry standards. The inspectors reviewed the personnel training and qualification process and the training and qualification records of five design and analysis authors and verifiers, and one contracted design and analysis author to verify conformance with the requirements in Criterion II, "Quality Assurance Program," of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel

Reprocessing Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." In addition, the inspectors discussed the personnel training and qualification process with Westinghouse management and technical staff. The attachment to this inspection report lists the documents reviewed by the inspectors.

b. Observations and Findings

The inspectors verified that training programs had been established and implemented for the indoctrination and training of personnel performing design and design verification activities to assure that proficiency was achieved and maintained. WEC 2.6, "Training," describes the policy and requirements for training and qualification of personnel performing activities affecting the quality of items and services supplied by Westinghouse. Specific training, education, and experience criteria for specific job functions are not defined in the procedure. Instead, WEC 2.6 identifies that the responsible manager determines the personnel competencies necessary for the assigned activities and assesses the associated needs for each activity. The manager ensures that necessary actions, including training, are completed to achieve the necessary competency. In addition, the responsible manager evaluates the actions taken to confirm that personnel are adequately trained, competent, and qualified to manage and perform assigned work activities. The manager identifies the applicable Westinghouse policies and procedures to which the employee is to be trained and documents them in the training needs matrix, which is maintained in the employees training file. The procedural training plan is communicated to the employee, the employee completes the training, and a record is maintained in the employees training file. The manager then reviews the file to ensure that the employee has completed the training on the Westinghouse policies and procedures.

The NRC inspection team interviewed Westinghouse staff to determine the process that Westinghouse managers use to ensure that their staff has achieved the necessary competency to perform assigned work. As described above, the employee's manager determines that an employee has achieved the necessary competency to perform assigned work. The responsible manager reviews the employee's education and experience and determines any additional formal training or on the job training that is necessary to perform assigned work. The identification of the additional training is not formally documented; however, the completion of the training and the performance of work are documented in quality records. The manager, in conjunction with the appropriate technical leads, reviews the employee's training records and completed work and determines which level of work the employee is competent to perform. For example, the manager will identify if the employee is competent to a) co-author a document with a more competent employee, b) independently author calculation notes provided the scope of work is within their area of competency, c) independently author and verify calculation notes within their areas of competency, or d) function as technical leads or subject matter experts that are qualified to oversee technical work in the areas identified and serve as reviewers to ensure competent staff work is being achieved. The outcome of this determination is documented in an uncontrolled qualification matrix. There is no controlled documentation attesting to this determination. Westinghouse took immediate corrective action and opened issue report number 13-094-M066 to address the lack of objective evidence on how different levels of qualification are achieved and documented. Since the process is documented and Westinghouse took immediate corrective action, this issue is considered of minor significance.

c. Conclusions

The inspectors concluded that Westinghouse's program requirements for training and qualification of personnel are consistent with the requirements of Criterion II of Appendix B to 10 CFR Part 50. The inspectors also concluded that Westinghouse's quality assurance manual and associated training and qualification procedures were adequate and effectively implemented. No findings of significance were identified.

3. Entrance and Exit Meetings

On April 3, 2013, the inspectors discussed the scope of the inspection with Mr. William Bedont, Westinghouse Process Manager, and with Westinghouse management and staff. On April 5, 2013, the inspectors presented the inspection results and observations during an exit meeting with Mr. Bedont and other Westinghouse management and staff. The attachment to this report lists the entrance and exit meeting attendees, as well as those interviewed by the inspectors.

On May 16, 2013, the inspectors conducted a subsequent exit meeting with Anthony Sicari, Principle Licensing Engineer, and other Westinghouse personnel to discuss a the identification of an additional nonconformance not presented at the April 5, 2013 exit meeting.

ATTACHMENT

1. ENTRANCE/EXIT MEETING ATTENDEES

Name	Title	Affiliation	Entrance	Exit	Interviewed
Richard McIntyre	Inspection Team Leader	NRC/NRO	X	X	
Samantha Crane	Inspector	NRC/NRO	X	X*	
Yuken Wong	Technical Specialist	NRC/NRO	X	X	
Edward Roach	Branch Chief	NRC/NRO		X*	
Joseph Colaccino	Branch Chief	NRC/NRO		X	
Bill Bedont	Resource Manager	Westinghouse	X	X	X
Mark Urso	Resource Manager	Westinghouse	X	X*	X
Richard Volomer	Fellow Engineer	Westinghouse	X	*	X
Gregory Meyer	Principal Engineer	Westinghouse	X	*	X
Gregory Imbrogno	Principal Engineer	Westinghouse	X	*	X
David Roarty	Consulting Engineer	Westinghouse	X	X	X
Matt Palamara	Senior Engineer	Westinghouse	X	*	X
Anthony Sicari	Principal Licensing Engineer	Westinghouse	X	X*	X
John Fasnacht	Director	Westinghouse	X	X	X
Doug Holderbaum	Director	Westinghouse		X*	X
Richard DeLong	Director	Westinghouse		X*	
Paul Russ	Director	Westinghouse		X	
Ron Wessel	Regulatory Inspection Coordinator	Westinghouse		*	
Jim Gresham	Regulatory Compliance Manager	Westinghouse		*	
Jim Brennan	Vice President Engineering Services	Westinghouse		*	
Terry Rudek	Director	Westinghouse		*	
Dick Whipple	Operations Manage	Westinghouse		*	
Bill Smoody	Principal Engineer	Westinghouse		*	
John Green	Licensing Manager	Westinghouse		*	

* denotes attendance at the May 16, 2013 exit meeting.

2. INSPECTION PROCEDURES USED

Inspection Procedure (IP) 43003, “Reactive Inspections of Nuclear Vendors.”

3. LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

The following items were found during this inspection:

Item Number	Status	Type	Description
99900404/2013-203-01	Open	NON	Criterion III
99900404/2013-203-02	Open	NON	Criterion XVI

4. DOCUMENTS REVIEWED

- APFIV-DRT-B21-MV006, Rev. 0, “AP1000 Scale Model Testing FIV Test Report”
- American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Article NG-3000, “Design,” Appendix I, Figure I-9.2.2, “Design Fatigue Curves for Austenitic Steels, Nickel-Chromium-Iron Alloy, Nickel-Iron-Chromium Alloy, and Nickel-Copper Alloy for $S_a \leq 28.2$ ksi, for Temperatures Not Exceeding 800°F,”1998 Edition
- APP-MI01-S3C-331, Rev. 3, “Flow-Induced Vibration (FIV) of the AP1000 Vortex Suppression Plate and Secondary Core Support Structures”
- APP-MI01-S3R-002, Rev. 3, “AP1000 Generic Reactor Vessel Internals (RVI) Summary Design Report”
- APP-RXS-M3C-029, Rev. 2, “AP1000 RVI RCP-Induced Vibration”
- APP-RXS-M3C-303, Rev. 0, “Calculation of Pump-Induced Pulsation Loads for AP1000 Reactor Vessel Internals”
- APP-MI01-S3C-107, Rev. 1, “AP1000 RESM – Flow-Induced Vibration Response Analysis”
- “Supporting Information for Westinghouse Flow-Induced Vibration Analyses of the AP1000 Reactor Vessel Internals”
- “Supporting RCP Response Analysis Sensitivity Study for Westinghouse Flow-Induced Vibration Analyses of the AP1000 Reactor Vessel Internals”
- Corrective Action Process (CAPs) Issue Reports (IR)
- CAPs IR 12-285-M038, October 2012, “Potential Damping Error in RCP FIV Analysis APP-RXS-M3C-029 R2 Appendix J”
- CAPs IR 12-286-W001, October 12, 2012, “Some loads were not specifically evaluated in the secondary core support structure analysis in AP1000 design analysis”
- CAPs IR 12-286-W003, October 12, 2012, “Formal documentation of a concern the analysis results in random turbulence analysis (APP-MI01-S3C-107) do not appear to be right”
- CAPs IR 12-286-W008, October 12, 2012, “Two quite different RVI reactor equipment system models (RESM) were used for AP1000 RVI design”
- CAPs IR 12-286-W009, October 12, 2012, “Formal documentation of a concern Random turbulence load needs to be evaluated in random vibration analysis (APP-MI01-S3C-107)”
- CAPs IR 13-045-M035, February 14, 2013, “Incorrect Phasing Relationship of Applied Force-Time Histories in AP1000 RVI FIV Turbulence Analysis”
- CAPs IR 13-094-M066, “Review and Assessment ARIDA Technical Qualification Process”
- LTR-ARIIDA-10-21, “MPR Associates Task-Specific QA Plan and ANSYS Installation Verification for AP1000 Reactor Vessel Internals (RVI) Analysis,” Revision 0, dated May 12, 2010
- QA Policy and Procedure Qualifications, SAP Data, dated April 4, 2013
- Regulatory Guide 1.20, “Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing,” Revision 2, issued May 1976WEC 2.6, “Training,” Revision 1, dated October 10, 2012
- WES-2011-155, Audit Package for MPR Associates, Inc., performed July 19-22, 2011

5. <u>INSPECTION QUESTIONS PROVIDED PRIOR TO THE ENTRANCE MEETING</u>

1. Identify all changes made to the AP1000 reactor internals since 2003
   a. Identify which changes, if any, are not included in the reactor equipment system models (RESM) for the reactor coolant pump (RCP) pulsation and the random turbulence (RT) analyses
   b. Describe how those changes were accounted for and any analyses or evaluations done to show that the current model is bounding (conservative)
2. The difference between the first core barrel beam mode in the RCP RESM and the AP1000 1/7th scale test is more than 10%. Describe and provide the justification that the RESM is adequate in predicting the AP1000 reactor internals dynamic response
3. Describe how you demonstrated that the dynamic results from the RESM models are conservative, and when the different modal frequencies are considered, the fatigue results are conservative
4. Describe the forcing function applied to the random vibration analysis model and the setup of the 1/7th scale test
   a. Describe how you demonstrated that the AP1000 reactor internals and the 1/7th scale model achieved similitude
   b. Describe the differences (e.g., forcing function, component layout, etc.) that caused the AP1000 core barrel displacements to be different from those in the historical test data
5. Explain the core barrel displacement behavior in the RT RESM
6. Describe the effect of the RCP pulsation loads on the energy absorbers and base plate
7. Discuss the impact to stress results when the RCP pulsation loads for all cases are included for the secondary core support structure (SCSS) columns
8. Describe how you demonstrated that applying random turbulence pressure loads in the upwards and downwards directions to the vortex suppression plate (VSP) is effectively doubling the loads
9. Describe how you confirmed that the upwards and downwards random turbulence pressure loads are applied in the most conservative manner (180 degrees out of phase).
10. Describe how you quantitatively demonstrated that existing stress results are conservative compare to the results if the random turbulence pressure load is applied to the base plate.
11. Describe and demonstrate how the reactor internals vibration predictions without the core shroud turbulence loads are conservative.
12. Provide the fatigue stress results of all reactor internal components and discuss how the results will be impacted if the assumptions (design changes, core barrel frequency, lower support plate displacement, RCP pulsation loads on SCSS, turbulence loads on VSP and base plate, turbulence loads on core shroud) are considered in the analyses.
13. Identify which ASME Code Section NG component stress limit curve (A, B, or C) was used for the reactor internals component vibration analyses.
14. Provide the date that the individuals who addressed the corrective action reports on the AP1000 reactor vessel internals vibration analysis obtained the various levels of flow-induced vibration qualification.
15. Describe the flow paths to the vessel head area and how the flow for the upper internals vibration preoperational test, which is conducted without the fuel assemblies, is conservative relative the flow expected during normal operations.
16. Describe how the reactor internals comprehensive vibration assessment program analysis meets RG 1.20 and identify which revision of RG 1.20 was used.

17. Describe how the computational fluid dynamics (CFD) calculation was benchmarked in accordance with Westinghouse procedures and practices. Describe any revisions made to the CFD calculation as a result of the benchmarking.