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U.S. Nuclear Regulatory Commission
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Your ref: Docket No. 52-006
Our ref: DCP/NRC2357

January 16, 2009

Subject: AP1000 Responses to Requests for Additional Information (SRP3)

Westinghouse is submitting a response to the NRC request for additional information (RAI) on SRP Section 3. This RAI response is submitted in support of the AP1000 Design Certification Amendment Application (Docket No. 52-006). The information included in this response is generic and is expected to apply to all COL applications referencing the AP1000 Design Certification and the AP1000 Design Certification Amendment Application.

Enclosure 1 provides the response for the following RAI:

RAI-SRP3.9.3-EMB2-07

Questions or requests for additional information related to the content and preparation of this response should be directed to Westinghouse. Please send copies of such questions or requests to the prospective applicants for combined licenses referencing the AP1000 Design Certification. A representative for each applicant is included on the cc: list of this letter.

Very truly yours,

D. A. Lindgren / for

Robert Sisk, Manager
Licensing and Customer Interface
Regulatory Affairs and Standardization

/Enclosure

1. Response to Request for Additional Information on SRP Section 3

D063
NRO

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ENCLOSURE 1

Response to Request for Additional Information on SRP Section 3

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP3.9.3-EMB2-07
Revision: 0

Question:

The staff conducted an on-site review of AP1000 component design on October 13 to October 17, 2008. The staff reviewed how Westinghouse translated DCD information into the design specifications for all components audited. The staff also reviewed the way in which Westinghouse documented the design analysis methodologies, criteria, and functional requirements in its design report for each major component in accordance with ASME Code, Section III. The staff requires response to the following Open Item in order to conclude its review of the proposed removal of the COL information item, currently addressed in the DCA.

Steam Generator - The transient resulting from primary system depressurization leading to external pressurization of the steam generator tubes leads to a calculated stress to allowable stress ratio of 0.96, and a fatigue limit ratio of 0.99. Since these ratios are close to their limiting values, the NRC audit team asked to examine the assumptions regarding the modeling of tube wall thinning.

The calculated stress on the tube wall depends on the wall thickness which depends on the integrated effects of erosion, corrosion, tube wear and chemical cleaning thinning. The rates of these phenomena are computed using correlations obtained from experimental simulations. Westinghouse is requested to provide confirmation that the margins to maximum stress and fatigue limits are adequate, considering the rates of erosion, corrosion, chemical cleaning and tube wear. Confirm that the experiments used to support evaluation of the erosion rate were performed under prototypic conditions, including flow velocity, and that the erosion rate used in the wall thickness calculations is conservatively derived from the experimental data.

Westinghouse Response:

The high fatigue ratio (0.99) in the tubes originates extensively from the 20 cycles of Inadvertent Reactor Coolant System (RCS) Depressurization umbrella transient, which is a stuck open pressurizer safety valve (PSV) starting from 100% power. Subsequent to this fatigue calculation, new inadvertent RCS depressurization transients were defined in which the 20 umbrella cycles were broken down into 5 cycles of a stuck open PSV and 15 cycles of inadvertent pressurizer spray. This division of the required 20 cycles is based on historical data which shows the majority of Level B RCS depressurization transients are events other than a stuck open PSV. The resulting fatigue usage in the tube analysis is expected to be below 0.90.

The calculation of the 0.96 tube collapse pressure limit ratio was also revised. The limiting inadvertent automatic depressurization system (ADS) transient was revised and the use of ASME Code Case N-759-1 was invoked in the calculation. The resulting tube collapse pressure limit ratio will be reduced to less than 0.90.

The margins for the calculated collapse limit and the fatigue usage are adequate, because the design basis tube wall thinning allowance is applied to the individual tubes. Satisfaction of the ASME Code criteria for these allowances assures the functional performance and integrity of the AP1000 tubes.

The tube wall thinning allowance for the Steam Generator (SG) tubes is derived from two sources. First, the Corrosion/Erosion allowance for the tubes inside and outside surface is based on data generated in WNEP-8661 Rev 1, "Corrosion /Erosion /Wear Allowances for the Steam Generator and Pressurizer Materials". This data was generated for the Inconel 600 tubes by reviewing available plant data and experimental results. From this report, it is concluded that 3 mils (0.003") total is a conservative value to represent the loss of material due to Corrosion/Erosion. A letter, NPD/E/PEN-99-097 "Stainless Steel and Inconel Material Loss", was issued stating that Inconel 690 (the material currently used) is more corrosion resistant than Inconel 600 and therefore, Corrosion/Erosion allowance calculated for Inconel 600 can be used for Inconel 690 conservatively. The use of 3 mils is also specified in Section 5.4.2.3.4 of the AP1000 Design Control Document.

Secondly, local tube wear due to Flow Induced Vibration (FIV) is calculated for local regions adjacent to the Anti-Vibration Bars (AVBs) and the Tube Support Plates (TSPs). This calculation is based on scaling measured workrate trends from full-sized U-bend tubes documented in CN-NCE-07-19 Rev 2, "Flow Induced Vibration and Tube Wear Evaluation for AP1000 Steam Generators". The analytical model developed to calculate limiting tube/AVB wear in the U-bend region uses empirical workrate trends obtained from prototypic test conditions that include nonlinear effects of the interactions at loose supports. The documented analyses cover the range of possible fit-up conditions determined by design and inspection of previous tube assemblies during and after fabrication. Workrates for both fluidelastic and turbulent excitation measured from tests are scaled to different steam generator operating conditions using appropriate fluid properties and vibration analysis parameters from qualified ATHOS models. Iterative calculations cover the 60-year operating period and account for increasing excitation with increasing wear in the case of fluidelastic excitation. These analysis methods provide a comprehensive review of the AP1000 tubes and demonstrate that the AP1000 SG has an adequate tube wall thinning allowance.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

Technical Report (TR) Revision:

None