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Your ref: Docket No. 52-006
Our ref: DCP/NRC2334

December 23, 2008

Subject: AP1000 Response to Request 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 the responses 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 RAIs:

RAI-SRP3.12-EMB-06

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,

Robert Sisk

Robert Sisk, Manager
Licensing and Customer Interface
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/Enclosure

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

D083
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-SRP3.12-EMB-06
Revision: 0

Question:

During the on-site technical review held the week of October 20, 2008, of ASME Class 1 piping for the AP1000 DCA, the NRC staff noted that the reactor coolant loop analysis did not couple the branch lines such as the surge line. Subsection 3.7.3.8.1 of DCD states that "if the ratio of the run piping outside diameter to the branch piping outside diameter (nominal pipe side) exceeds or equals 3.0, the branch piping can be excluded from the analysis of the run piping." Several branch lines do not meet this ratio, therefore, should be included in the RCL piping analysis. The staff requests that the applicant explain this discrepancy with DCD and take action to address this DCD conformance issue.

Westinghouse Response:

The branch piping has been excluded from the reactor coolant loop analysis because the criteria in Subsection 3.7.3.8.1 do not apply to the hot and cold leg piping. Just as attached piping is excluded from primary equipment models, the branch piping of the surge line, automatic depressurization system stage 4 (ADS4), RNS suction line, and several smaller lines are excluded from the analysis of the hot and cold leg piping.

The reactor coolant loop (APP-RCS-PLA-050) is unique in that the stiffness and mass characteristics are closer to that of equipment than a typical piping analysis package. With the relatively short run length, comparatively large pipe diameter and pipe thickness, both the cold and hot legs have much less flexibility than a typical run length of pipe. The large interplay of the hot and cold leg piping with the reactor pressure vessel and steam generator extends the boundary of the piping analysis package to include primary equipment as well as primary loop piping.

No non-conformance exists because the reactor coolant loop piping is treated as a rigid piece of equipment (fundamental frequency greater than 33 Hz) and not a flexible pipe.

Reference(s):

None

Design Control Document (DCD) Revision:

Sections 3.7.3.8.1 and Appendix 3C are revised as shown below.

3.7.3.8.1 Supporting Systems

This subsection deals with the analysis of piping systems that provide support to other piping systems. The methods used for the analysis of the primary loop piping are described in Appendix 3C. [*The supported piping system may be excluded from the analysis of the supporting piping system when the ratio of the supported pipe to supporting pipe moment of inertia is less than or equal to 0.04.*

If the ratio of the run piping outside diameter to the branch piping outside diameter (nominal pipe size) exceeds or equals 3.0, the branch piping can be excluded from the analysis of the run piping. The mass and stiffness effects of the branch piping are considered as described below.

Stiffness Effect

The stiffness effect of the decoupled branch pipe is considered significant when the distance from the run pipe outside diameter to the first rigid or seismic support on the decoupled branch pipe is less than or equal to one half the deadweight span of the branch pipe (given in ASME III Code Subsection NF).

Mass Effect

Considering one direction at a time, the mass effect is significant when the weight of half the span (from the decoupling point) of the branch pipe in one direction is more than 20 percent the weight of the main run pipe span in the same direction. Concentrated weights in the branch pipe are considered. A branch pipe span in x direction is the span between the decoupled branch point and the first seismic or rigid support in the x direction. A main run pipe span in the x direction is the piping bounded by the first seismic or rigid support in the x direction on both sides of the decoupled branch point. Similarly, the same definition applies to the spans in other directions (y and z).

If the calculated branch pipe weight is less than 20 percent but more than 10 percent of the main run pipe weight, this weight is lumped at the decoupling point of the run pipe for the run pipe analysis. This weight can be neglected if it is less than 10 percent of the main run pipe weight.

Required Coupled Branch Piping

If the stiffness and/or mass effects are considered significant, the branch piping is included in the piping analysis for the run pipe analysis. The portion of branch piping considered in the *analysis adequately represents the behavior of the run pipe and branch pipe. The branch line model ends in one of the following ways:*

First six-way anchor
Four rigid/seismic supports in each of the three perpendicular directions
Rigidly supported zone as described in subsection 3.7.3.13.4.2]*

APPENDIX 3C

REACTOR COOLANT LOOP ANALYSIS METHODS

The AP1000 reactor coolant loop (RCL) model consists of three-dimensional finite elements such as pipes, beams, elbows, masses, and springs. The structural model is subjected to internal pressure, thermal expansion, weight, seismic, and pipe break loadings with imposed boundary conditions. The finite element displacement method is used for the analysis. The stiffness matrix for each element is assembled into a system of simultaneous linear equations for the entire structure. This set of equations is then solved by a variation of the Gaussian elimination method, known as the wave-front technique. This technique makes it possible to solve systems of equations with a large number of degrees of freedom using a minimum amount of computer memory.

3C.1 Reactor Coolant Loop Model Description

The piping model of the reactor coolant loop consists of a number of elements of given dimensions, sizes, and physical properties that mathematically simulate the structural response of the physical system. The system model contains the reactor pressure vessel (RPV), two steam generators (SGs), four reactor coolant pumps (RCPs), the reactor coolant loop piping, and the primary equipment supports. A two-loop model is developed for the AP1000 reactor coolant loop system.

The stiffness and mass effects of branch piping connected to the primary loop piping are considered when significant (subsection 3.7.3.8.1).

3C.1.1 Steam Generator Model

3C.1.1.1 Steam Generator Mass and Geometrical Model

The steam generator is represented by discrete masses. The geometry of the steam generator vessel is used to determine the properties of the equivalent piping elements that join the steam generator masses. The modulus of elasticity and coefficient of thermal expansion corresponding to the thermal conditions are applied to the steam generator equivalent piping elements.

3C.1.1.2 Steam Generator Supports

The values of the steam generator support stiffnesses and locations of the supports are determined from the finite element models of the support members. The stiffness of the upper lateral supports includes the steam generator shell flexibility. The local concrete building flexibility is included in the support stiffness.

3C.1.2 Reactor Coolant Pump Model

3C.1.2.1 Static Model

The reactor coolant pump is modeled using equivalent pipe elements. The modulus of elasticity and thermal expansion coefficient corresponding to each thermal condition are applied to these pipe elements.

3C.1.2.2 Seismic Model

The reactor coolant pump is represented by a super element to represent the mass and stiffness characteristics of the pump. The reactor coolant pump model is a detailed model similar to that used to qualify the pump.

3C.1.2.3 Reactor Coolant Pump Supports

There are no reactor coolant pump supports. Two reactor coolant pumps are attached to the steam generator channel head in each of the reactor coolant loops.

3C.1.3 Reactor Pressure Vessel Model

3C.1.3.1 Mass and Geometrical Model

The reactor pressure vessel model consists of equivalent pipe, stiffness, and mass elements. The elements represent the vessel shell, the vessel core barrel, the fuel assemblies, and the integrated head lift package.

The reactor pressure vessel is modeled with equivalent pipe elements and connecting stiffnesses. The equivalent pipe element properties of the vessel and barrel are those of the cylindrical structures. The beam properties of the reactor internals are adjusted to simulate their fundamental frequency. The appropriate modulus of elasticity and coefficient of thermal expansion are used for the equivalent pipe elements representing the reactor pressure vessel.

3C.1.3.2 Reactor Pressure Vessel Supports

The reactor pressure vessel is supported at the four reactor pressure vessel inlet nozzles. Each support consists of a vertical stiffness and a lateral tangential stiffness. The support is represented by a stiffness matrix. The reactor pressure vessel supports are active for the analyzed loading conditions. The reactor pressure vessel model includes the effects of the vessel shell flexibility at the inlet and outlet nozzles. The local concrete building flexibility is included in the support stiffness.

3C.1.4 Containment Interior Building Structure Model

A containment interior building structure finite element model is not required because the seismic inputs to the reactor coolant loop model are provided at all of the building attachments to the reactor coolant loop.

3C.1.5 Reactor Coolant Loop Piping Model

The reactor coolant loop piping model consists of piping elements and bends. Each reactor coolant loop has two cold legs and one hot leg. The straight runs and bends of the cold leg and hot leg are input with the nominal dimensions. Each reactor coolant loop branch connection is represented by a node point. The reactor coolant loop piping model contains ~~a~~ distributed masses of the hot and cold leg piping for static deadweight analysis and lumped masses representing the hot and cold leg piping for dynamic analysis.

3C.2 Design Requirements

The reactor coolant piping is qualified to the requirements of the ASME Code, Section III, Subsection NB, 1989 Edition with 1989 Addenda.

The loadings for ASME Code, Section III, Class 1 components are defined in subsection 3.9.3. The following loadings are considered in the reactor coolant loop piping analysis:

- Design pressure (P)
- Weight (DW)
- Thermal expansion during normal operating condition
- Thermal expansion during other transient conditions (not part of this appendix)
- Safe shutdown earthquake (SSE)
- Design basis pipe break (DBPB)

- Building motions due to automatic depressurization system sparger discharge into the IRWST

- Thermal stratification during transient conditions

In addition to the analyses of these loads, the reactor coolant piping is analyzed for the effect of cyclic fatigue due to the design transients and earthquakes smaller than SSE.

3C.3 Static Analyses

3C.3.1 Deadweight Analysis

The reactor coolant loop piping system is analyzed for the effect of deadweight. The deadweight analysis is performed without considering the dry weight of the directly supported equipment. The effects of the auxiliary branch piping on the reactor coolant loop are generally negligible by the design of the auxiliary supports. A deadweight analysis is performed to include the total weight of the reactor coolant loop piping and the water weight in the components.

The reactor coolant loop deadweight model includes the corresponding active reactor coolant loop supports - reactor pressure vessel supports, and the steam generator column and lower and intermediate lateral strut supports. The steam generator upper lateral snubber supports are considered as inactive.

3C.3.2 Internal Pressure Analysis

The effects of the internal primary coolant pipe pressure are used in the calculations of forces and moments for both the reactor coolant loop piping and equipment supports.

3C.3.3 Thermal Expansion Analysis

The reactor coolant loop piping is analyzed for the effects of thermal expansion. The thermal expansion analysis model considers the expansion of the reactor coolant loop piping, reactor pressure vessel, steam generator, reactor coolant pump, and the equipment supports. The stiffness effects of the auxiliary piping on the reactor coolant loop expansion are generally negligible by the design of the auxiliary lines supports.

3C.4 Seismic Analyses

The reactor coolant loop piping is analyzed for the dynamic effects of a safe shutdown earthquake (SSE).

The model used in the static analysis is modified for the dynamic analysis by including the lumped mass characteristics of the piping and equipment. The effect of the equipment motion on the reactor coolant loop piping and support system is obtained by modeling the mass and stiffness characteristics of the equipment in the overall system model. The reactor coolant loop seismic analysis is performed at normal full-power operation. This operating condition is considered based on the lower probability of occurrence of the earthquake at reactor coolant loop temperatures below full power.
The time history integration method of analysis is used for the reactor coolant loops. The seismic input considers the soil profiles described in subsection 3.7.1. This input is obtained from the nuclear island seismic analysis. The duration of the input is between 12 to 20 seconds, depending on the duration needed to envelop the design response spectra. For each of the soil profiles, either the building stiffness is varied by + or - 30 percent, or the time scale is shifted by + or - 15 percent, to account for uncertainties. Rayleigh damping is used with the loop components at 4 percent of critical damping. The steam generator snubbers have different stiffnesses in tension and compression. The mean value of the tension and compression stiffness is used in order to keep the model linear. The reactor pressure vessel vertical supports are acting downward only and are preloaded by deadweight, pressure, and thermal expansion loadings. A verification is performed that no lift-off of the vessel occurs at the location of these supports.

3C.5 Reactor Coolant Loop Piping Stresses

To prevent gross rupture of the reactor coolant loop piping system, the general and local primary membrane stress criteria must be satisfied. This is accomplished by satisfying Equation (9) in paragraph NB-3652 of the ASME Code, Section III. The secondary stress caused by thermal expansion is qualified by satisfying Equation (12) in paragraph NB-3653 of the ASME Code, Section III.

3C.6 Description of Computer Programs

This section provides a list of computer codes used for the AP1000 reactor coolant loop system analysis. Brief descriptions of the functions of each computer code are the following:

WECAN/WECAN-PLUS – Performs Structural Analysis Using Finite Element Analysis Method. WECAN is a mainframe program while WECAN-PLUS uses a workstation. Displacements and loads are calculated at the pipe elements, supports and equipment nozzles for pressure, deadweight, thermal, and seismic loadings.

STRESCAL – Post-processes the WECAN output data to calculate time history loads in selected elements. Input consists of a Modal Force File and a time history Modal Coefficient File for each mode. STRESCAL combines the results for the modes.

ANSYS – Performs Structural Analysis Using Finite Element Analysis Method. ANSYS is used in the loop model described in subsection 3.7.2. 3.1 and shown in Figure 3.7.2-7. This ANSYS model may be modified in accordance with this Appendix and ANSYS may be used in lieu of WECAN for the reactor coolant loop qualification analysis.

PRA Revision:

None

Technical Report (TR) Revision:

None