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 for sections of the steam generator above the tubesheet. For the steam generator channel head, a super element is used to represent the stiffness characteristics that link the steam generator lower shell with the steam generator supports and nozzles. 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 include 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 represented by a super element to represent the mass and stiffness characteristics of the pump. For a thermal expansion analysis, rigid links are modeled in parallel with a super element with the thermal expansion coefficient incorporated.

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 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. The moment stress due to pressure is considered negligible for the ASME Code pipe stress equations.

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 with time history input generated from the enveloped basemat response spectra of the soil cases described in subsection 3.7.1. The duration of the input is between 12 to 20 seconds, depending on the duration needed to envelop the design response spectra. Three runs were performed based on the envelope of the soil profiles, the building model at nominal stiffness, and at stiffness varied by + or - 30 percent to account for uncertainties. The reactor coolant loop uses separate time history displacement input from the building analysis at the primary support locations. Full direct integration is used with Rayleigh damping for 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. The time history analysis is performed to evaluate the effect of lift-off of the vessel 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:

ANSYS – Performs Structural Analysis Using Finite Element Analysis Method. Displacements and loads are calculated at the pipe elements, supports, and equipment nozzles for pressure, deadweight, thermal, and seismic loadings.