APPENDIX 3C

REACTOR COOLANT LOOP ANALYSIS METHODS

The AP600 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 and seismic 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 containment interior building structure, the reactor coolant loop piping, the surgeline piping, and the primary equipment supports. A two-loop model is developed for the AP600 reactor coolant loop system.

The containment interior building structure model is included in the seismic system model when the time-history integration method is used.

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 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 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 multi-node model. The reactor coolant pump casing and motor are represented by reduced mass and stiffness matrices for horizontal and vertical motion. The reactor coolant pump rotor is represented by vertical mass and stiffness elements. The simplified reactor coolant pump model is obtained from a detailed model of the reactor coolant 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 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 bellows that place a given stiffness in series with a rigid piping element. 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

The containment interior building structure finite element model is made up of three-dimensional beam elements, spar elements, and pipe elements. This simplified building model is correlated to a detail model of the building.

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 mass for static deadweight analysis and lumped masses for dynamic analysis.

3C.2 Design Requirements

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

The containment interior concrete is represented by a nominal Young's modulus, including the effect of material uncertainty. The value of the modulus is chosen to provide frequencies close to the significant equipment frequencies.

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 lateral strut supports. The steam generator upper lateral snubber and bumper supports are considered as inactive. The containment interior building structure model is not considered in the deadweight analysis.

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 with a coupled model of the reactor coolant loops and the interior concrete building. 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 four 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. Composite modal damping is used with the building components at 5 percent and the loop components at 4 percent of the critical damping. The equipment support nonlinearities at the steam generator upper lateral snubbers and the reactor pressure vessel vertical supports are included in the coupled model. The steam generator snubbers have different stiffnesses in tension and compression. The reactor pressure vessel vertical supports are acting downward only and are preloaded by deadweight, pressure, and thermal expansion loadings.

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 AP600 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.