DYNAMIC ANALYSIS OF PRIMARY REACTOR COOLANT SYSTEM SUBJECTED TO SEISMIC AND LOCA LOADS, INCLUDING SHIELD-BUILDING INTERACTION

D.T. RAMANI

Nuclear Engineering Department, Motor-Colombus Consulting Engineers Inc., CH-5401 Baden, Switzerland

SUMMARY

This paper is intended to describe the development of a unique method of analysis for a primary reactor coolant piping system to evaluate its adequacy and a need for structural support system optimization under hypothetical seismic and loss-of-coolant-accident loading conditions. An integrated approach is presented for a linear three-dimensional dynamic analysis of a typical 900 MWe PWR nuclear reactor coolant loop, accommodating dynamic-coupling and interaction effects of shield-building including soil-foundation springs and damping parameters. Also included are major influences on sensitive equipment due to their variable support boundary conditions and rigid restraints. Due to vital nature of reactor coolant system, important dynamic design criteria have to be established in accordance with estimating natural frequencies, damping characteristics and behaviour of the overall system during LOCA and strong motion earthquakes as well as to optimize the structural design of various components.

With this procedure, complex major reactor coolant components, such as piping, reactor vessel, control rod drives, steam generators, nozzles, elbows, pumps, pressurizer, support-restraints, snubbers and shield-building are systematically idealized as equivalent beam elements and lumped masses and synthesized into a multi-degree-freedom three-dimensional lumped-mass mathematical model in which global stiffness matrix, $[K]$, and mass matrix, $[M]$, for the entire system are established. Each mass point is assigned six degrees of freedom system and to economize, dynamic degrees of freedom of each mass point are reduced according to dynamic characteristics of the coolant system. The equations of motion of such a lumped-mass mathematical model under any forcing function, $\{F(t)\}$, may be expressed as; $[M]\{\ddot{X}\} + [C]\{\dot{X}\} + [K]\{X\} = \{F(t)\}$, in which $[C]$ is a damping matrix and $\{X\}$ is a displacement vector. Many refinement features are introduced for an accurate behaviour of the model under such complex loading conditions as; LOCA, seismic, deadweight, thermal expansion, transient and sustained mechanical loads.

For a three-dimensional seismic analysis, response-spectrum form of input in preference to time-history input is considered due to computational economy. However, in LOCA analysis, a forcing-function time-history input at selected points is used. A sophisticated computer code, such as modified PIPDYN-II has been utilized to obtain natural frequencies mode-shapes absolute and relative displacement, absolute accelerations, internal forces and moments as well as stresses based on RMS criteria. Special attention is paid to modes of close-frequencies, and their participation-factors. Modal damping is evaluated using Biggs' weighted energy method. Both LOCA and seismic conditions are combined to evaluate final stresses to be compared with ASME-III Code allowable to accommodate optimum changes if any. Confidence level is gained from safety margins over computed stress patterns. Numerical results are furnished for a typical reactor coolant system subjected to both seismic and LOCA load analyses.

1. Introduction

The importance of knowledge of seismic and LOCA (loss-of coolant accident) load dynamic responses of primary reactor coolant piping system is related to the safety and operational requirements in assessing the adequacy, structural integrity and trouble-free functioning of each of the system components. In order to provide protection against the remote possibility of uncontrolled release of radioactive material into atmosphere, it is imperative that class-I structures, such as coolant system of a nuclear power plant, are designed to resist both seismic disturbance and LOCA loads. In addition, the primary coolant structural support system must be optimized to meet the extreme situation of combined hypothetical seismic and LOCA loading conditions. In earlier stages [1,2,3] of the nuclear power plant design, due to limitations on digital computer codes, various components of the primary coolant system were seismically analysed by specifying equivalent static 'g' (gravity) loads in both horizontal and vertical directions and then locally combining with critical LOCA loads, without regard to participations of such vital components as reactor vessel including control-rod drive mechanism, steam generators, coolant pumps, pressurizer, core-flooding tanks, snubbers, shield-building and support-restraint structures. In the past, dynamic analysis [4,5,6] of piping systems have been characterized as being very involved and complex to exactly account for the elasto-dynamic properties of various participating components and their unpredictable behaviours in oscillating under the influence of seismic or LOCA load disturbances. However, with the advent of modern computer codes, such as PIPDYN-II [7], SAP IV [8] STARDYNE [9] etc. and better insight into the dynamic behaviour of each component of the coolant system, an integrated system, consisting of vital equipment together with nuclear power piping and structural support system, is analysed conservatively as a three-dimensional assemblage. With the present sophisticated knowledge from tests [10,11,12,20,21] it is concluded that the physical and dynamic characteristics of vital equipment such as reactor vessel, steam-generator, pump etc., under the influence of strong motion earthquake [13,14] are very similar to power piping and that local shell action is negligible; permitting an analyst to idealize the system as a lumped-mass dynamic model. Earlier, the interaction of piping and equipment [13,14,19] with the shield-building was also ignored and treated separately by floor response-spectrum method, under the assumption that the mass of primary coolant system of a typical PWR plant was extremely small in comparison to that of the shield-building. However, with subsequent testing [10, 12,15,17] it is confirmed that such a system does behave dynamically as one unit, when subjected to strong-motion earthquake or any other excitation. One major objective of this paper is to present an outline of important steps involved in an accurate analysis of primary coolant system subjected to both seismic and LOCA excitations by performing meaningful (16) Class-I evaluation.

2. Method of LOCA Analysis

For loss-of-coolant conditions, the loads acting on the supports and restraints of the reactor coolant system components will generally be calculated using the three-dimensional loop mathematical model previously described. For all breaks, the LOCA thrust time-histories (shown in Figs. 5 to 8) will be applied at the appropriate locations near the components of interest and then break-locations will be chosen such that the worst possible loads will be generated on the component supports. For guillotine breaks, the appropriate pipe elements will be mathematically modified and the thrust forcing functions applied to determine the dynamic response. A total of 5 guillotines and 8 longitudinal splits are considered in the system dynamic analysis. The pipe-break loads are usually based on the full cross-sectional area of pipe for both longitudinal and double-ended ruptures. However, the orientation of the forces around the circumference of the pipe as well as their location along the pipe, was arbitrary although the critical locations are determined by trial and error. Several locations and orientations may be necessary to ensure that all support components are subjected to the maximum postulated break-load, accounting for greater number of longitudinal splits than that of guillotines. The loads imposed on equipment-support system were assumed to be limited by the load-carrying capacity of the pipe and the calculations for the moment capability of the pipe included the shearing effects due to transverse loads. For longitudinal splits, the force perpendicular to the pipe is usually evaluated such that plastic bending moment capabilities of a particular pipe would not be exceeded based on static considerations. Since it is theoretically possible to plastically yield the pipe under a shear load, the minimum break-length for developing full cross-sectional area of pipe is conservatively assumed to be one pipe-diameter. Also the pipe properties of the break area are conservatively assumed to be equivalent to those of an unbroken pipe. These important assumption introduce bounds on the maximum shearing loads acting on the equipment - support system and moment-shear interaction curves are then generated to ensure that maximum possible loads acting on the supports and restraints. This was concluded from results in Figs. 17 to 20.

3. Mathematical Background

For linear systems, the resulting equations of motion are usually treated by modal analysis and theoretically it is possible to describe the entire coolant system by one model to evaluate its response by either response-spectra method or time-history approach. In this investigation, response due to seismic input is obtained by using response-spectrum technique because of computational economy, while transient LOCA analysis was performed by using time-history approach since time-dependent forcing-functions for various pipe-breaks are digitally available. Both methods are available in PIPDYN-II [7] and SAP IV [8]

FIG. 1 MODEL FOR PRIMARY COOLANT LOOP INTERACTING WITH SECONDARY SHIELDING WALL

FIG. 2 MATHEMATICAL MODEL OF REACTOR COOLANT PIPING ASSEMBLY IN ELEVATION

FIG. 3 MATHEMATICAL MODEL OF REACTOR-COOLANT SYSTEM SHOWING REACTOR VESSEL PUMP AND COLD-LEG IN ELEVATION

FIG. 4 FINITE ELEMENT ANALYSES OF T-JOINT & PIPE-SHELL INTERSECTION

FIG. 7 TIME-HISTORY LOCAL THRUST LOAD ON THE STEAM GENERATOR DUE TO COLD-LEG PIPE BREAK

FIG. 8 TIME-HISTORY LOCAL THRUST LOAD ON THE STEAM GENERATOR DUE TO HOT-LEG PIPE BREAK

FIG. 5 TIME-HISTORY LOCAL THRUST LOAD ON THE REACTOR VESSEL DUE TO HOT-LEG DOUBLE-ENDED RUPTURE

FIG. 6 TIME-HISTORY LOCAL THRUST LOAD ON REACTOR VESSEL DUE TO COLD-LEG DOUBLE-ENDED RUPTURE

codes in which desired seismic and LOCA analysis for the coolant system are easily carried out, using lightly damped (0.5%-1%) values for various components.

3.1 Time-History Modal Analysis for LOCA Forcing Functions

For a multi-degree of freedom system subjected to time-dependent forcing function, the coupled equations of motion in matrix form is:

$$[m_s]\{\ddot{q}\} + [C]\{\dot{q}\} + [K_s]\{q\} = \{F(t)\} \quad (3)$$

Where $[m_s]$ =diagonal mass matrix, $[C]$ =damping coefficients matrix, $[K_s]$ =system stiffness matrix, $\{\ddot{q}\}$, $\{\dot{q}\}$ and $\{q\}$ are column matrices of accelerations, velocities and displacements at the coordinates of the system respectively and $\{F(t)\}$ =a forcing function. The natural frequencies, ω , and mode-shapes, $\{\phi\}$, of the undamped system of 'n' degrees of freedom may then be obtained from the following equation:

$$([K_s] - \omega_n^2 [m_s])\{\phi_n\} = \{0\} \quad (4)$$

In order to decouple eq. (3), a transformation to normal coordinates is accomplished by substituting, $\{q\} = \{\phi\}\{y\}$ into eq. (3), which then yields:

$$[m_s]\{\phi\}\{\ddot{y}\} + [C]\{\phi\}\{\dot{y}\} + [K_s]\{\phi\}\{y\} = \{F(t)\} \quad (5)$$

Pre-multiplying above equation by $\{\phi\}^T$ and using the orthogonality condition, yields:

$$\{\phi\}^T [m_s] \{\phi\} \{\ddot{y}\} + \{\phi\}^T [C] \{\phi\} \{\dot{y}\} + \{\phi\}^T [K_s] \{\phi\} \{y\} = \{\phi\}^T \{F(t)\} \quad (6)$$

In numerical computations, it is usually convenient to normalize the mode shapes $\{\phi\}$ with respect to $[m_s]$, resulting in the following relations:

$$\{\phi\}^T [m_s] \{\phi\} = [I], \{\phi\}^T [K_s] \{\phi\} = [\Delta], [C_\xi] = \{\phi\}^T [C] \{\phi\} \quad (7)$$

where $[I]$ = the identity matrix, $[\Delta]$ is a diagonal matrix of eigenvalues ω_n^2 and $[C_\xi]$ is a diagonal matrix such that $[C_\xi]_{ii} = 2\xi\omega_i$, in which $\omega_i = (\Delta_i)^{1/2}$ and ξ is the estimated critical damping ratio. Since $[C_\xi] = 2\xi[\Delta^{1/2}]$ and using the relation $[C_\xi] = \{\phi\}^T [C] \{\phi\}$, the matrix $[C]$ may be manipulated in the form,

$$[C] = 2\xi [m_s] \{\phi\} [\Delta^{1/2}] \{\phi\}^T [m_s] \quad (8)$$

and readily computed. Equation (6) may be rewritten in normal coordinates as:

$$\{\ddot{y}\} + [C_\xi]\{\dot{y}\} + [\Delta]\{y\} = \{Q(t)\} \quad (9)$$

Where $[C_\xi]$ and $[\Delta]$ are previously defined and $\{Q(t)\} = \{\phi\}^T \{F(t)\}$. The forcing-function, $\{Q(t)\}$ is a modified vector obtained from known digitized time-history LOCA loads. The matrix formulation in eq. (6) may be written as n decoupled equations of motion in the form:

$$(\ddot{y}_r + 2\xi\omega_r \dot{y}_r + \omega_r^2 y_r) = Q_r(t) \quad (10)$$

$r=1, 2, 3, \dots, n$

The solution to eq. (10) can be readily obtained by Laplace Transform and is given as:

$$y_r(t) = \frac{1}{\omega_r} \int_0^t \frac{e^{-\xi\omega_r(t-\tau)}}{\sqrt{1-\xi^2}} \sin[\omega_r \sqrt{1-\xi^2}(t-\tau)] Q_r(\tau) d\tau \quad (11)$$

Which for a lightly damped system ($\xi \leq 5\%$) can be approximated by the relation:

$$y_r(t) = \frac{1}{\omega_r} \int_0^t e^{-\xi \omega_r(t-\tau)} \sin[\omega_r(t-\tau)] Q(\tau) d\tau \quad (12)$$

Once LOCA time-history loads, $Q(\tau)$, are simulated into the system, dynamic response can be easily obtained by numerical integration.

3.2 Response Spectra Techniques

The seismic response for a multi-degree of freedom system can be computed by the response-spectrum-technique, in which case the coupled equations of motion in the matrix form is:

$$[m_s]\{\ddot{q}\} + [C]\{\dot{q}\} + [K_s]\{q\} = -\ddot{S}_a(t)[m_s]\{D\}_j \quad (13)$$

Where $\ddot{S}_a(t)$ = seismic time-history support motion and $\{D\}_j$ = column matrix governing seismic motion in j th direction. Once eigenvalues, ω_n , and eigenvectors, $\{\phi\}$, are obtained from the eq. (4), then using transformation into normal coordinates and premultiplying by $\{\phi\}^T$ the eq. (13) gives:

$$\{\phi\}^T [m_s] \{\phi\} \{\ddot{y}\} + \{\phi\}^T [C] \{\phi\} \{\dot{y}\} + \{\phi\}^T [K_s] \{\phi\} \{y\} = -\ddot{S}_a(t) \{\phi\}^T [m_s] \{D\}_j \quad (14)$$

Using normalising and orthogonality conditions yields:

$$\{\phi\}^T [m_s] \{\phi\} = [M], \{\phi\}^T [K_s] \{\phi\} = [\omega_n^2] [M], \{\phi\}^T [C] \{\phi\} = [2\xi_n \omega_n] \quad (15)$$

in which $[M]$ = the diagonal generalized mass matrix and ξ_n = the fraction of critical damping. Substituting eq. (15) into Eq. (13) and then multiplying both sides by $[M]^{-1}$ and defining $\{\Gamma_n\}_j = [M]^{-1} \{\phi\}^T [m_s] \{D\}_j$, yields:

$$\{\ddot{y}\} + [2\xi_n \omega_n] \{\dot{y}\} + [\omega_n^2] \{y\} = -\ddot{S}_a(t) \{\Gamma_n\}_j \quad (16)$$

Where $\{\Gamma_n\}_j$ is termed as the column matrix of participation factors in j th direction. Eq. (16) indicates that each mode of vibration corresponds to a single-degree-of-freedom system and therefore maximum modal responses due to an earthquake (defined by $\ddot{S}_a(t)$) can be evaluated using response-spectrum concept. The equivalent modal damping, D_n , used in the modal response analysis, can be evaluated for each mode on the basis of weighted average damping (17), as discussed previously. The absolute maximum modal accelerations, $\{\ddot{a}_n\}$, can then be obtained in the matrix form as:

$$\{\ddot{a}_n\} = \{\phi\} [S_\omega] [\Gamma] \quad (17)$$

in which $[S_\omega]$ = diagonal matrix of maximum modal spectral accelerations and $[\Gamma]$ = diagonal matrix of participation factors. Maximum modal relative displacement, $\{\delta_n\}$, and maximum modal inertia forces, $\{P_n\}$, can be evaluated as:

$$\{\delta_n\} = \{\ddot{a}_n\} [V\omega^2], \{P_n\} = [m_s] \{\ddot{a}_n\} \quad (18)$$

To obtain maximum modal moments and shears, inertia forces, $\{P_n\}$, are applied on the system and the complete response at any node point in the multi-degree-of-freedom system can be obtained by combining on the basis of square root of the sum of squares (SRSS) method.

4. Numerical results, discussions and conclusions

A typical 900 MWe PWR primary coolant system is analysed herein and subjected to both site seismic disturbances and LOCA load time-histories to evaluate the adequacy and integrity of various components in accordance with ASME Section

III and ANSI-B31.1 codes. Modified PIPDYN-II and STARDYNE computer codes are used to perform the dynamic and other types of stress analyses in order to evaluate compliance of final stress levels with NB-3600. The mathematical model of the entire coolant system consists of 592 members (both straight and curved) and 516 node points (both structural and mass joints) with each mass joint having three translational degrees of freedom. The definition of the mathematical model is input in the form of actual section properties, masses, joint-coordinates, connectivities and parameters of the shield building system.

4.1 Seismic Analysis

Seismic loadings on the coolant system contribute rather extensively to the final combined stress levels of the coolant system. Seismic inputs for this nuclear plant consist of both safe-shut-down earthquake (SSE) as well as operational basis earthquake (OBE). These are intended to represent respectively maximum probable earthquake and an upper limit of potential hazard. Corresponding response spectra were developed from three known recorded time-histories, namely the El Centro, Taft and Golden-Gate. Response spectra are then normalized to the local site base acceleration intensity of 0.15g (SSE) and 0.75g (OBE), as illustrated in Fig.21. Through the use of digital computer codes, necessary dynamic parameters for the model, such as flexibility matrix, mass matrix, natural frequencies, mode-shapes and participation factors are computed. First 80 frequencies for this mathematical model are shown in Table-1. Based on the participation factors and the frequencies, significant modes are selected for seismic acceleration inputs from the response-spectra. Typical analyses consist of separate inputs for X,Y and Z Earthquake directions.

Although various components of the model possess different critical damping values; that is: shield-wall-5% to 7% reactor, steam generators, pumps pressurizer and core-flooding tanks 1% to 1.5% and piping - 0.5% to 1%, the conservative critical damping values of 0.5% for OBE and 1% for SSE for all components were assumed in the analysis. The shield-building and its foundation are assumed to be very rigid as to permit the ground motion to be directly transmitted to the system anchors and supports. Soil-structure interaction effects transmitted to the pedestal are assumed small and neglected. Presented in Table-1 are the natural frequencies of first 80 modes of vibration of the system, while Table-2 shows partially composite modal damping values for an OBE acting horizontally along X-axis of the reactor-coolant system. For the critical effects to occur in the steam-generator, the model was assumed to be cold and the steam-generator was disconnected at the upper-support from the shield-wall. Similar studies were made on other coolant components.

4.2 LOCA Load Analysis

In order to comply with safety criteria, it is usually recommended to combine

extreme loading for each component of the coolant system with other types of loading. Therefore, in accordance with Appendix-F of ASME-III code, which regulates the evaluation of faulted condition, the intent of coolant design is to limit the primary and peak stresses resulting from postulated LOCA condition when combined with operation basis earthquake (OBE) condition. Although nature of the problem is such that the system under LOCA loading is highly non-linear (time-dependent elasto-plastic analysis), the analysis herein is limited conservatively to linear elastic deformations and then to simplify the evaluation of dynamic responses with unknown time-histories at points of pipe-break. The flexibility characteristics of the complete model reflect the location and nature of break. For longitudinal splits, it is conservatively assumed the pipe properties in the vicinity of the break were same as for the unbroken pipe to ensure maximum transmission of loads to the supports and restraints. Since the first five natural frequencies did not vary significantly for various breaks considered, the major contribution to the stiffness of the system comes from the supports.

For reactor-vessel thrust due to postulated double-ended rupture in the hot-leg, the leak-area is restrained to 3.0 ft^2 whereas for cold-leg double-ended break, corresponding leak area is restrained to 8.5 ft^2 . The time of rupture in each case is restricted to 160 milliseconds period. Identical values are also assigned to the steam-generator hot-leg and cold-leg pipe break leak-areas; however, the duration of rupture is restricted to 80 and 160 milliseconds respectively. Time-history LOCA Thrust load on the pressurizer due to surge-line break is not so critical. The duration of rupture is about 16 seconds and the leak-area is restrained to 0.45 ft^2 . In each time-history dynamic analysis, uniform modal damping of 1% is assumed and the response of the structural system is evaluated for the specified duration after the postulated occurrence of pipe rupture. The dynamic response consists of nodal displacements, acceleration, loads and support reactions. An examination of participation factors and the significance of number of modes used in the time-history analysis indicated that dynamic response could be adequately approximated with the first 50 to 60 modes of vibration. Figure 18 shows the horizontal displacement of the steam generator top-head due to hot-leg guillotine. The vertical displacement is relatively much smaller. Maximum horizontal displacement occurs at approximately 0.07 seconds after break. Figure 19 illustrates the horizontal displacement at the top of the coolant pump, indicating that the maximum response occurs at approximately 0.055 seconds after break. For the steam generator, the foundation moment due to LOCA load is much higher than that due to seismic load alone. The results are based on a zero gap connection between the generator and upper level lateral support. This is a valid assumption when the generator is hot and expanded against the support sys-

FIG. 9 — DISTRIBUTION OF REACTOR FOUNDATION SHEAR FORCE DUE TO HOT-LEG RUPTURE

FIG. 10 — DISTRIBUTION OF REACTOR FOUNDATION MOMENT DUE TO HOT-LEG PIPE RUPTURE

FIG. 11 — DISTRIBUTION OF REACTOR FOUNDATION SHEAR FORCE DUE TO COLD-LEG PIPE RUPTURE

FIG. 12 — VARIATION OF REACTOR VESSEL FOUNDATION MOMENT DUE TO COLD-LEG PIPE BREAK

TABLE-1
EIGENVALUES AND NATURAL FREQUENCIES OF THE PRIMARY REACTOR COOLANT SYSTEM INCLUDING THE INTERACTION EFFECTS OF SECONDARY SHIELD WALL NATURAL FREQUENCIES

EIGENVALUES	MODE NO.
0.3512134E-02	64
0.2685568E-01	65
0.1781778E-02	66
0.3787288E-01	67
0.3950979E-01	68
0.16287307E-02	69
0.15504531E-02	70
0.74681122E-03	71
0.54488323E-03	72
0.38608859E-03	73
0.17481678E-02	74
0.21345119E-03	75
0.20608512E-03	76
0.19124061E-03	77
0.18036737E-03	78
0.17349764E-03	79
0.16181581E-03	80
0.15730191E-03	81
0.14950265E-03	82
0.13615374E-03	83
0.85077956E-04	84
0.67941388E-04	85
0.6056678E-04	86
0.54252460E-04	87
0.49937015E-04	88
0.45717055E-04	89
0.41569301E-04	90
0.40278640E-04	91
0.39093244E-04	92
0.37111653E-04	93
0.35753821E-04	94
0.27800679E-04	95
0.26147129E-04	96
0.25708560E-04	97
0.23568303E-04	98
0.23020612E-04	99
0.15279517E-04	100
0.1329517E-04	101
0.91766509E-05	102
0.85242439E-05	103
0.85056264E-05	104
0.81652667E-05	105
0.72874413E-05	106
0.65912323E-05	107
0.62192518E-05	108
0.62198367E-05	109
0.60974023E-05	110
0.51900659E-05	111
0.40207818E-05	112
0.38766981E-05	113
0.37203266E-05	114
0.34364095E-05	115
0.19303857E-05	116
0.17458766E-05	117
0.16223549E-05	118
0.14280399E-05	119
0.12422443E-05	120
0.11698022E-05	121
0.10565466E-05	122
0.85282455E-06	123
0.78275799E-06	124
0.76297859E-06	125
0.72797292E-06	126
0.69510277E-06	127
0.66228246E-06	128

tem. For the LOCA thrust analysis, much higher critical loads on the shield-wall occurred for hot-leg double-ended rupture than for any other case. A double-ended rupture of the pump suction-line at the lower end induced large vertical excitations of the pump and they were reduced by stiffening the pump supports.

4.3 Conclusions

Results of the seismic and LOCA analyses indicate the complex nature of the problem and confirm that component support and restraint loads based on static analysis considerations are inadequate for design. It is also specifically concluded that much higher foundation and support loads occur under LOCA conditions than for seismic. Higher foundation loads also occurred on both the reactor and steam generator for hot-leg break than for cold-leg break. The contribution to component member loads from cross-coupling interaction effects is not predicted from static analysis and similarly interaction effects in the dynamic analysis are more pronounced when all the components are included. It is also of interest to note that the seismic stresses are quite small as compared to those generated from LOCA. The seismic stresses are nevertheless significant when combined with other types of stress levels. Present results have shown that for the resulting combined loads, the supports and restraints for all the components remain functional assuring the safety criteria. However, in accordance with ASME-III code, certain joints failed to satisfy Equation-10 ($3 S_m$ criteria) and certain joints failed to meet the Equation-11 and cumulative damage criteria. Similarly for faulted condition, using OBE condition, the primary stresses are below the allowables for all joints.

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