Evaluation of Stress Histories of Reactor Coolant Loop Piping for Pipe Rupture Prediction

S.C. Lu, R.A. Larder

Lawrence Livermore National Laboratory, University of California, P.O. Box 808, L-90, Livermore, California 94550, U.S.A.

A.L. Chan

Science Applications Inc., 5 Palo Alto Square, Suite 200, Palo Alto, California 94304, U.S.A.

S.M. Ma

EG & G, Inc., P.O. Box 204, San Ramon, California 94582, U.S.A.

SUMMARY

This paper describes the analyses used to evaluate stress histories in the primary coolant loop piping of a selected four-loop pressurized water reactor nuclear power station. The results of the analyses were used as input to a simulation procedure for predicting pipe rupture probabilities of the reactor coolant system presented by another SMiRT-6 paper J6/6. In order to make the simulation as realistic as possible, best estimates rather than conservative assumptions were considered throughout. The best estimate solution, however, was aided by a sensitivity study to assess the possible variation of outcomes resulted from uncertainties associated with these assumptions.

Sources of stresses considered in the evalution were pressure, dead weight, thermal expansion, thermal gradients through the pipe wall, residual welding, pump vibrations, and finally seismic excitations. The best estimates of pressure and thermal transient histories arising from plant operations were based on actual plant operation records supplemented by specified plant design conditions. Seismic motions were generated from response spectrum curves developed specifically for the region surrounding the plant site.

Stresses due to dead weight and thermal expansion were computed from a three dimensional finite element model which used a combination of pipe, truss, and beam elements to represent the reactor coolant loop piping, the reactor pressure vessel, coolant pumps, steam generators, and the pressurizer. Stresses due to pressure and thermal gradients were obtained by closed form solutions.

Seismic stress calculations considered the soil structure interaction, the coupling effect between the containment structure and the reactor coolant system. A time history method was employed for the seismic analysis.

Calculations of residual stresses accounted for the actual heat impact, welding speed, weld preparation geometry, and pre- and post-heat treatments. Vibrational stresses due to pump operation were estimated by a dynamic analysis using existing measurements of pump vibrations.

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1. Introduction

One of the major efforts of the Load Combination Program conducted at Lawrence Livermore National Laboratory in 1980 was to estimate the probability that a large loss-of-coolant accident (LOCA) and an earthquake occur simultaneously [1]. This information will be considered by the U.S. Nuclear Regulatory Commission (NRC) to reassess the requirement to combine earthquake and LOCA loads in the nuclear power plant design. The approach considered a numerical procedure which simulates weld joint fracture of the primary coolant loop piping due to fatigue crack growth. The numerical simulation requires the evaluation of pipe stresses arising from all relevant loads during the plant life. The required stress analysis is presented in this paper.

2. Approach

Figure 1 is a flow diagram showing how the stresses are calculated. Since the objective of this work is to perform a realistic simulation, best estimate loads and material properties are used for the calculations whenever possible. If such data are unavailable, conservative values for loads and properties are used instead. Many parameters, such as damping, support stiffness,and soil properties, are varied to estimate the sensitivity of the results to those parameters. Sources of stresses considered in the evaluation are pressure, dead weight, thermal expansion, thermal gradient through the pipe wall, residual welding, pump vibrations, and finally seismic excitations.

Since the results of the stress analysis will be used as input to a computer program for simulation of fatigue crack growth, the stress evaluation is subject to some requirements. Instead of merely predicting the maximum stress and the stress range for fatigue it is necessary to characterize the complete variation in stress in all girth butt welds in the piping for the plant lifetime. The fatigue crack growth simulation requires both the steady state or quasi-static stresses as well as the cyclic variation in the stresses. In the cases where the stress distribution is not uniform through the pipe wall thickness the stress calculation requires description of this distribution. The variation in stress around the circumference of the pipe is handled by calculating the stress of 48 positions by a stress post processor. The stress post processor also calculates a single numerical factor used to characterize the entire stress history of a particulat seismic event.

3. Plant Description

Unit 1 of the Zion Nuclear Power Station (Zion Unit 1) was selected as the demonstration plant for this study. Zion Unit 1 is a four-loop pressurized water reactor (PWR) plant. The primary components of the four-loop PWR nuclear steam supply system (NSSS) of Zion Unit 1 are the reactor vessel, the four steam generators, and the four reactor coolant pumps. The primary loop piping, which includes the hot leg (reactor pressure vessel to steam generator), the crossover pipe (steam generator to coolant pump), and the cold leg (reactor coolant pump to the reactor pressure vessel) of each of the four loops, provides the basis for the current study. These pipes have outside diameters of 30 inches or more, and have walls that are approximately 2.5 inches thick.

The primary piping, nozzles, and fittings are made of various grades of cast and wroght 316 stainless steel. In this study, no attempt has been made to differentiate the mechanical properties in these different components. The ASME code requirements for the minimum specified room temperature yield and ultimate strengths are 30 ksi (207 MPa) and 70 ksi (483 MPa), respectively. The code allowable stress at the operating temperature ranges from 11.8 ksi (81 MPa) to 16.8 ksi (116 MPa), depending on whether piping fittings or nozzle material are considered.

4. Non Seicmic Stresses

4.1 Stresses Due to Pressure, Thermal Expansion, and Dead Weight

A simple formula was used to compute the stresses due to internal pressure [2]. The thermal expansion and dead weight stresses were computed using the SAP4 computer program [3]. The pipe, reactor pressure vessel, steam generators, reactor coolant pumps, pressurizer, valves and component supports are represented by a combination of pipe, beam and truss elements. All four loops are modeled in the analysis. The input temperatures for the thermal expansion analysis are $588^{O}F$ for the hot leg, and 540^{O} for the cold leg and crossover leg. For the surge line, the average temperature between the hot leg and the pressurizer ($635^{O}F$) is used. The operating pressure is 2250 psi. The maximum axial stresses in the pipe under these conditions were found to be 16.2 ksi, 8.4 ksi, and 10.4 ksi, respectively, in the hot leg, crossover, and cold leg.

4.2 Thermal Gradient Stresses

During the thermal transient condition, the changes in fluid temperature create a nonuniform temperature distribution and, therefore, stress variations across the pipe wall. For clarity, these stresses are referred to as thermal radial grandient stresses. The thermal radial gradient stresses are evaluated for the hot leg, cold leg, and surge line. Because of the similarity in temperature, the resulting thermal gradient stresses for the cold leg are also used to evaluate the crossover leg. Temperatures as well as stresses through the pipe wall were calculated analytically [2]. Typical stress profiles in the hot leg produced by the steam line break from no load transient are shown in Fig. 2. The evaluation considers the following thirteen plant transients:

Reactor heatup
Reactor cooldown
Plant loading, 5% per minute
Plant unloading, 5% per minute
10% step load decrease
10% step load increase
Large step decrease in load
Loss of load from full power

Loss of offsite power

Loss of flow in one loop (the last loop)

Loss of flow in one loop (the other loops)

Reactor trip from full power

Steam line break from zero power

4.3 Vibrational Stresses

Vibration stresses in the piping system are generally caused by vibration of the rotating equipment, by flow-induced vibration (such as vortex shedding, cavitation, or flashing), or by fluid pressure pulses. The importance of vibration stresses depends on the stress magnitude. Becuase there may be many cycles, even moderate vibrational stresses can be detrimental to the piping system.

To predict vibration stresses, we performed dynamic time history analyses and compared the results to the preoperatinal test data from the Indian Point Nuclear Plant [4] , which has the same thermal output and system configuration as Zion Unit 1. Flow-induced vibration was not included in this study. A simple harmonic forcing function based on the rotating mass, eccentricity, and rotational speed of the reactor coolant pump was used for the time history analysis. The effect of the upper restraints on the steam generator was also investigated. The time history analysis was performed using the SAP4 code with 3% of critical damping. The bending mode of the steam generator was found to be the first mode. Its modal frequency was 7.2 Hz with the upper restraint included and 2.4 Hz with the upper restraint excluded; both values agreed well with the test results. On the other hand, calculated deflections were much lower than those found during preoperational testing. To account for this difference, the resulting stresses were scaled by the ratio of the measured to the analytical deflections. The resulting stresses were very low (i.e., a few hundred psi) and did not influence the fatigue growth.

4.4 Resudual Stresses

Residual stress values are computed using a finite element thermal stress analysis model [2] . The calculation accounts for the actual heat impact, welding speed, weld preparation geometry, and pre- and post-heat treatments. Of particular interest are the axial residual stresses since the observed mode of pipe cracking is predominantly circumferential. Stresses in the cold leg are shown in Fig. 3. Stresses are compressive at or near the inside surface. This is probably due to the great thickness of the pipe. The distribution of the axial stress in the pipe indicates that the contribution of residual stresses to fatigue crack growth is negligible.

5. Seismic Stresses

5.1 Best Estimate Analysis

Best-estimate, state-of-the-art techniques have been used to estimate the seismic stresses at the weld joints of the primary loop piping. The analytical model for the seismic stress analysis includes a reactor coolant loop model, reactor building model, and a soil model. The reactor model includes all major components affecting the seismic response of a four-loop PWR-1 Westinghouse nuclear steam supply system

(NSSS)--the components, supports, and the interconnecting piping. The major components include one reactor pressure vessel, four steam generators, four reactor coolant pumps, and one pressurizer. Major piping includes the hot leg, crossover, and cold leg. This model is a reduced version of the NSSS model used for the nonseismic stress analysis. A detailed description of this model is given in Ref. 5. The two components of the reactor building are the prestressed concrete containment shell and the reactor building internal structures. These two structures share a common foundation; otherwise, they are structurally separated. The containment shell is represented by a 3-D, 12-mass stick model. Beam elements connect the lumped masses. The internal concrete structure is represented by a 3-D finite element model composed mainly of plate elements to simulate concrete walls and slabs. The NSSS model is connected to the internal concrete structure mainly at the operating floor and at the foundation level.

The dynamic soil stiffness parameters (shear modulus and material damping) required for the soil-structure interaction analysis were estimated using a site-specific soil model as input to the SHAKE computer program [6]. The soil was idealized into layers of constant soil properties which include unit weight, layer thickness and description, low-strain shear modulus, and shear wave velocity values.

Seismic inputs were generated for a variety of peak accelerations from one time OBE to five times SSE. For each given peak acceleration, an earthquake with a specific spectrum and duration was randomly selected from a base of 5000 potential earthquakes at the Zion site. Artifical accelerograms were generated using the computer program SIMOKE [7]. Three statistically independent artifical accelerograms were generated from each target spectrum for two horizontal and one vertical direction, respectively. The ratio of peak vertical acceleration to peak horizontal acceleration was established based upon an analysis of earthquake-motion data.

The soil-structure interaction analysis was carried out using the CLASSI code [8]. The formulation of CLASSI is generally categorized as the substructure technique (or 3-step solution technique). The soil-impedance function matrices and structural dynamic (modal) properties were separately determined. The structural modal properties, in this case, were determined by SAP4. Steam generator sway was found to dominate the first few modes. Reactor coolant pump motion became important in the higher modes. The dynamic soil stiffness and damping input to CLASSI for establishing the impedance function were taken from the SHAKE analysis. The output from CLASSI is the three-dimensional moment-time history at the selected primary piping joints.

Dead weight and thermal expansion moments were combined with the seismic response to evaluate the fatigue damage and maximum stress. The fatigue damage calculation is based upon a Paris-type fatigue crack growth relation. The fatigue damage factor and maximum stress are computed at 48 locations around the circumference of the joints under investigation, and the maximum value is used in the crack growth analysis. Fig. 4 shows the maximum axial stress in the hot leg to the pressure vessel joint as it increases with the peak horizontal ground acceleration.

5.2 <u>Sensitivity Study</u>

The results of the best-estimate indicates that seismically induced stresses are small compared to normal operating streses. A limited sensitivity study was performed to evaluate the variability of these results with changes to damping, seismic input, soil-structure interaction, and dynamic analysis method. Compared to the results using the best-estimate input parameters, we found that a reduction in damping, use of the NRC Regulatory Guide 1.60 design spectra [9] , and elimination of soil-structure interaction effects (i.e., fixed base) can all significantly increase the joint stresses.

Use of a two-step response spectrum analysis technique, as compared to the simple one-step analysis employed in the best-estimate study, also increased the stresses by over a factor of two. The two-step method is inherently conservative since it used the enveloped floor response spectrum curve; whereas, the majority of the supports are near the foundation level. Separate seismic inputs occurring at the different supports could provide the most accurate analysis; however, this is beyond the current scope.

The stresses obtained in this analysis were significantly less than those predicted by Westinghouse for the design of the NSSS [10] . By inputing conservatively low values for support stiffnesses and using the R. G. 1.60 response spectrum as used by Westinghouse, we were able to reproduce the original design results. The low value of stresses calculated using the best-estimate input parameters appears to be a more realistic estimate of the actual piping stresses during a seismic event.

6. Conclusions

The primary coolant loop piping of Zion Unit 1 PWR plant was analyzed for stresses arising from internal pressure, thermal expansion, dead weight, thermal gradient in pipe wall, residual welding, vibrations, and seismic events. Stresses are primarily induced by pressure, thermal expansion, and dead weight. Thermal gradient contributes the nonuniform stress through the pipe thickness. Vibrations produced low stresses (a few hundred psi) and residual welding induced compressive stresses at or near the inner surface of the pipe. Both are considered insignificant from the point of fatigue crack growth.

Seismic excitations at Zion Site also produced low stresses in the primary loop pipe according to the best-estimate analysis. The results of the sensitivity study led to the conclusion that the degree of conservatism in normal seismic stress design calculations is very high. Consequently, the low value of stresses calculated using the best-estimate input parameters appears to be a more realistic estimate of the actual piping stresses during a seismic event.

7. References

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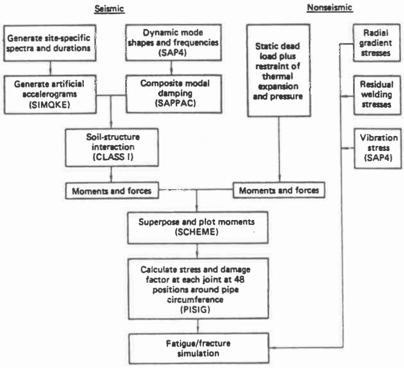

Fig. 1 Flow Chart of the Stress Calculations

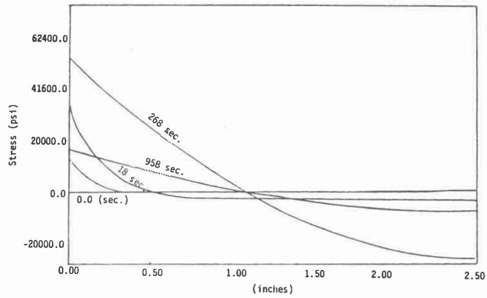

Fig. 2 Stress Profiles in Hot Leg Due to Steam Line Break from No Load Transient

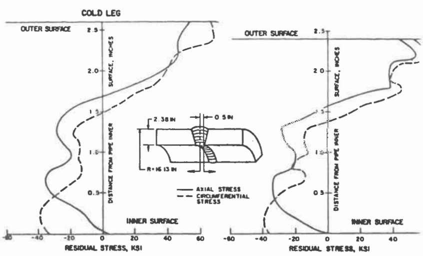

Fig. 3 Stresses Due to Welding to Cold Leg.

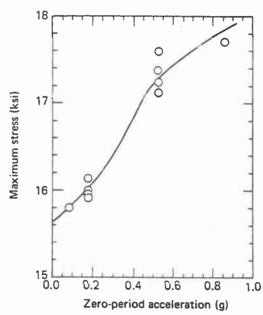

Fig. 4 Maximum Axial Stress in Hot-leg-to-pressure-vessel joint.