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Revision 1

SAN ONOFRE NUCLEAR GENERATING STATION UNITS 2 AND 3

SNUBBER REDUCTION PROGRAM FOR NUCLEAR PIPING SYSTEMS

Prepared for

Nuclear Regulatory Commission

Prepared by

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1.0 INTRODUCTION

Recent operating experience of nuclear power plants has highlighted reliability and maintenance problems associated with the use of snubbers on nuclear power plant piping systems (references 1, 2, and 3). Additionally, concerns have been expressed by knowledgeable industry personnel that piping systems may be overly designed with respect to seismic and other postulated plant design events (i.e., high energy line break), at the expense of optimum plant design for normal operating events. Over design of plant piping results in large numbers of unnecessary seismic supports (struts and snubbers), pipe whip restraints, and jet impingement barriers. Pipe supports, restraints, and barriers not needed to provide adequate design margins for seismic and other plant design events can restrict free thermal expansion of the pipe, hinder inservice inspection (ISI) of piping welds and components, and increase occupational radiation exposure (ORE) to plant personnel.

Recently completed testing at ANCO laboratories (reference 4) and by Teidoguchi (reference 5) have also demonstrated that piping systems can be subject to earthquake excitations at least three to five times larger than permitted by the ASME Code without failure of the piping. The ANCO testing also demonstrated that failure of a snubber or support in the Z-bend system tested did not result in loss of pressure retaining integrity of the piping. Historical data of fossil power plants and process plants subject to large earthquakes have not found failures of any major piping systems (references 6 and 7). In addition, examination of snubbers on the decay heat removal system at Three Mile Island Unit 2 found most to have failed as a result of a severe fluid dynamics transient. However, the piping system still retained its pressure retaining integrity (reference 8).

Other recently completed testing of typical nuclear piping systems has also demonstrated that snubbers may actually increase piping system response to the seismic event. This testing, sponsored by the Electric Power Research Institute (EPRI) at the University of California at Berkeley, concluded that "the influence of snubber connections on the dynamic response of the piping

appears to be unpredictable, although generally higher accelerations, forces in the connections and strains in the piping were introduced" (reference 9).

These concerns of piping over design for the seismic event have resulted in both industry and regulatory (reference 10) investigations into analytical methods, design criteria, and industry practices used for piping design, fabrication, and installation.

Southern California Edison Company (SCE) has been closely following these activities as principally conducted by the Nuclear Regulatory Commission (NRC), Pressure Vessel Research Committee (PVRC), American Society of Mechanical Engineers (ASME), and EPRI. Results of these activities to date have demonstrated that the seismic design of San Onofre Units 2 and 3 piping is excessively conservative and that many of the seismic supports (principally snubbers) are neither needed nor desirable to provide adequate design margins for the seismic events. Removal of these unnecessary seismic supports would result in piping systems of increased reliability, lower plant maintenance costs, and decreased occupational radiation exposure (ORE) to plant personnel.

Additionally, nuclear power plant operating experience, recent test data, and analytical advances in the mechanisms of High Energy Line Break (HELB) have demonstrated that the HELB design criteria of San Onofre 2 and 3 is also excessively conservative. Although SCE is not proposing to remove any pipe whip restraints or jet impingement barriers at this time, SCE is proposing modification to HELB criteria that will facilitate removal of unnecessary snubbers. However, SCE is reviewing information associated with the elimination of arbitrary intermediate pipe breaks (and associated pipe whip restraints) at the Catawba, Votgle, Byron, and Braidwood nuclear power plants and may apply for an associated criteria change at a future date.

Scope of the snubber reduction program is presented in Section 2.0. Design criteria for piping, pipe supports, and in-line components is presented in Section 3.0. Analytical methodology that will be utilized for the seismic event is presented in Section 4.0.

2.0 SCOPE

Scope of the snubber reduction program includes all safety related plant piping systems, snubbers on the Control Element Drive Mechanisms (CEDM), but exclusive of the nuclear steam supply system (NSSS) piping equal to or greater than 30 inches in diameter. Also excluded are snubbers used for seismic support of anchored equipment such as the reactor pressure vessel, steam generators, and reactor coolant pumps.

3.0 DESIGN CRITERIA

Licensing commitments as provided in the San Onofre 2 and 3 Final Safety Analysis Report (updated FSAR) will be followed except as specifically noted below.

3.1 Piping

All non NSSS piping will continue to be analyzed to the 1974 edition, summer 1974 addenda, of the ASME Boiler and Pressure Vessel Code, Section III (reference 11). NSSS piping will continue to be analyzed to the 1971 edition of the ASME Section III Code. In special circumstances, later editions and addenda may be used provided that the new requirements are reconciled with the original design requirements as provided in Section XI of the ASME Code.

3.1.1 Design Allowables

All piping loads, load combinations, and allowable stresses identified in the San Onofre 2 and 3 updated FSAR (sections 3.9.3 and 3.9.4) will be complied with.

3.1.2 High Energy Line Break Criteria

Design criteria for selection of postulated locations of High Energy Line Break (HELB) is provided in Section 3.6.2 of the FSAR. This criteria is essentially that of Standard Review Plan 3.6.2 (reference 12a) and Regulatory Guide 1.46 (reference 13), and provides for the selection of HELB locations at terminal ends and at two or more intermediate locations of highest stress. Evolution of these criteria has been summarized in NUREG-1061 (reference 14b) which concludes that current nuclear power plants have too many pipe whip restraints and jet impingement barriers for the extremely low probability HELB event.

Revision 0 of Standard Review Plan 3.6.2, the current revision of Regulatory Guide 1.46, and the updated San Onofre 2 and 3 FSAR imply that each time the piping system is reanalyzed and the stress pattern changes, the intermediate locations of postulated HELB must be revised. Such revision can result in the relocation of existing pipe whip restraints and jet impingement barriers, or the addition of new restraints and barriers. However, restraint and barrier modifications are not desirable from considerations of the extremely low probability of a HELB, ORE of plant personnel, further hinderance of ISI of the piping and components, hinderance of equipment maintenance, added potential to restrict free thermal expansion of the piping, and plant economics.

As discussed in reference 15, it was the intent of the NRC in issuing revision 1 to Standard Review Plan 3.6.2 that postulated arbitrary intermediate locations of HELB not be revised each time the pipe stress analysis is changed (as used herein arbitrary intermediate breaks are those postulated to provide a minimum of two breaks between anchors. An example are breaks of ASME Class 2 piping with equation 9 and 10 stress below 0.8 $(1.2\ S_h + S_A)$).

However, SCE proposes to maintain all existing postulated locations of HELB and not adding any new locations of HELB regardless of changes to the pipe stress pattern. This position is supported by the following:

1. The ANCO testing (reference 4) clearly demonstrated that piping systems have capacities to withstand excitations at least three to five times that equivalent to ASME Service Level D allowables without violating pressure retaining integrity. In addition, the testing demonstrated that at least some support failure could be withstood. Thus, exceeding arbitrary intermediate break stress criteria, but maintaining stresses below Code allowables, would not compromise piping integrity or capability to withstand the SSE.

2. The additional removal of snubbers that this position would provide will not change normal operating stresses. In addition, it eliminates the risk of inadvertent snubber lockup.

3. The removal of additional snubbers that this position would allow will result in increased space access for ISI of piping and components, increased space access for equipment maintenance, and decreased ORE of plant personnel.

3.2 Piping Supports

3.2.1 ASME Class 1, 2, and 3 Pipe Supports

Load combinations, applicable Code edition/addenda, and allowable stresses for NSSS and non NSSS pipe supports are provided in Sections 5.4 and 3.9 respectively of the San Onofre Units 2 and 3 updated FSAR. In special circumstances, later editions and addenda of the ASME Code may be used provided that the new requirements are reconciled with the original design requirements as provided in Section XI of the ASME Code.

The effects of both thermal anchor motion and seismic anchor motion will be considered as primary loads for evaluation of pipe supports and auxiliary steel.

3.2.2 ASME Welds

Weld allowables for ASME Class 1, 2, and 3 pipe supports will continue to be that provided by the applicable edition/addenda of Section III of the ASME Code.

3.2.3 Auxiliary Steel

For auxiliary steel, whose design loads are determined to be greater than current design loads, the load combinations and allowable stresses will

be provided by Standard Review Plan 3.8.4 (reference 12b) as limited by the following:

1. The allowable limit for compression members during emergency and faulted condition loads will be 1.33S.

2. For angle members, the unsupported length divided by the member thickness shall be less than 270.

3. For auxiliary steel subject to eccentric load application, uniform torsional stress, warping shear stress, and warping normal stress will be determined and included for comparison with appropriate allowables.

3.3 Mechanical Equipment

Load combinations, allowable stresses, and applicable Code versions identified in Sections 3.9 and 5.4 of the updated San Onofre 2 and 3 FSAR will be met for mechanical equipment. This includes vessels, pumps, heat exchangers, tanks, valves, and containment penetrations.

4.0 METHODOLOGY

Methodology that will be utilized by SCE in performing the snubber reduction program will include one or more of the following:

4.1 Response Spectra Method

Piping response to the inertia portions of the seismic input may be determined by either the uniform method or the Independent Support Motion (ISM) method. ISM is also referred to as multiple support motion or multiple input response spectra analysis in the literature. Acceptable analytical techniques include those defined in Appendix N of the 1983 edition of ASME Section III.

For the ISM method, each pipe support will be identified with a group experiencing a common response spectra and common seismic anchor movement (pseudo static component of the seismic response) according to its attachment point to the structural (building) model.

For the uniform method, the response spectra of all applicable pipe supports will be enveloped to determine the response spectra to be utilized for the entire piping subsystem. Each piping support will also be identified with the applicable seismic anchor movement.

4.1.1 Closely Spaced Frequencies

As demonstrated and recommended in NUREG/CR-3811 (reference 16) modal and directional responses will be combined by the square root of sum of the squares (SRSS) method without considering closely spaced frequencies. This is consistent with the design practices of non-NSSS San Onofre 2 and 3 piping (FSAR Section 3.7B.5). For NSSS piping, in the original design, closely spaced modes were combined by the absolute sum method as described in FSAR section 3.7.3.7.

As an alternate to the SRSS method, the Complete Quadratic Combination procedure defined in reference 17 may be used.

4.1.2 Sequence of Combinations

As demonstrated and recommended in NUREG/CR-3811 (reference 16), any sequence can be selected for combining modal and spacial components.

4.1.3 Combination of Group Responses for Independent Support Motion (Inertia Contribution)

For the ISM method, responses from each of the pipe support groups will be combined by the absolute sum method.*

4.1.4 Combination of Group Responses (Anchor Motion Contribution)

Per the recommendations of reference 16, pseudo static movement of each pipe support group will be combined by the absolute combination procedure to determine the seismic anchor movement response, as is the current design practice of San Onofre Units 2 and 3.

4.1.5 Combination of Directional Components

Results from each direction of earthquake (X, Y, and Z directions) will be combined by the SRSS method to determine total seismic response. This is consistent with the current design practices of San Onofre Units 2 and 3 piping (FSAR section 3.7B.5).

* Pending completion of the current Brookhaven study on ISM utilizing PVRC damping (follow on work to reference 16), SCE may apply to combine responses from each of the pipe support groups by the SRSS method.

4.1.6 Combination of Inertia and Anchor Motion Contributions

Seismic design loads on pipe supports and reaction loads on in-line components will be determined using a SRSS combination of inertia and anchor motion contributions. This procedure is in accordance with the recommendations of the Brookhaven study.

For evaluation of pipe stress, the seismic inertia contribution will continue to be classified as a primary load and seismic anchor motion will continue to be classified as a secondary load. This is consistent with the current design basis of San Onofre 2 and 3 piping.

4.2 Time History Method

As an alternate to the response spectra method defined in Section 4.2, piping system response may be determined by the time history method. Inertia and anchor motion contributions will be combined by the SRSS method, as discussed in Section 4.1.6.

4.3 Damping

Seismic response spectra analysis will be based on the damping curve developed by the PVRC in WRC Bulletin 300 (reference 18) and accepted by ASME for nuclear piping as Code Case N-411 (reference 19).

For piping frequencies in the range of 10 Hz to 20 Hz, a linear interpolation of 2% and 5% response spectra is considered an acceptable approximation of the response spectra which would be generated by utilizing the PVRC frequency variable damping.

For time history analysis, damping values provided in Table 3.7-22 of the San Onofre 2 and 3 FSAR will be used. These are identical to those recommended in Regulatory Guide 1.61 (reference 20).

4.4 Peak Shifting

To account for uncertainties in the natural frequencies of the piping system, either the peak broadening procedures of reference 21 or the peak shifting procedures of references 18 and 22 may be used. For the case of peak shifting, shifting will be performed about a minimum of two peaks of the response spectra.

5.0 REFERENCES

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4. Electric Power Research Institute, NP-3746, "Dynamic Response of Pressurized Z-Bend Piping Systems Tested Beyond Elastic Limits and With Support Failures", December 1984.

5. H. Teidoguchi, "Experimental Study on Limit Design for Nuclear Power Plant Facilities During Earthquakes, 1973", JPNRSR-5 (USERDA Technical Information Center); Part 2.2, "Vibration Tests of the Distribution Piping Systems".

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9. University of California at Berkeley, EPRI Project RP-364-8, "The Results of an Experimental Program on the Seismic Response of Multiply Supported Piping Systems", Final Report, February 1984.

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12. U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants", July 1981.
    a. Standard Review Plan 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping".
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    b. Volume 3, "Evaluation of Potential for Pipe Breaks", November 1984.
    c. Volume 4, "Evaluation of Other Dynamic Loads and Load Combinations", December 1984.
    d. Volume 5, "Summary - Piping Review Committee Conclusions and Recommendations", April 1985.

15. U.S. Nuclear Regulatory Commission, "Catawba Nuclear Station Unit 2, Safety Evaluation for the Elimination of Arbitrary Intermediate Pipe Breaks", Docket No. 50-414, March 1984.

16. Brookhaven National Laboratory, NUREG/CR-3811, "Alternate Procedures for the Seismic Analysis of Multiply Supported Piping Systems", August 1984.

17. E. L. Wilson, et.al., "A Replacement for the SRSS Method in Seismic Analysis", Earthquake Engineering and Structural Dynamics, Volume 9, 187-192 (1981).

18. Welding Research Council, Bulletin 300, December 1984.

19. ASME Code Case N-411, "Alternative Damping Values for Seismic Analysis of Piping Section III, Division I, Class 1, 2, and 3".

20. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants", October 1973.

21. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.122, "Development of Floor Design Response Spectra for Seismic Design for Floor-Supported Equipment or Components", Revision 1, February 1978.

22. ASME Code Case N-397, "Alternative Rules to the Spectral Broadening Procedures of N-1226.3 for Classes 1, 2, and 3 Piping".