UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

July 2, 1979

IE Bulletin No. 79-14

SEISMIC ANALYSES FOR AS-BUILT SAFETY-RELATED PIPING SYSTEMS

Description of Circumstances:

Recently two issues were identified which can cause seismic analysis of safety-related piping systems to yield nonconservative results. One issue involved algebraic summation of loads in some seismic analyses. This was addressed in show cause orders for Beaver Valley, Fitzpatrick, Maine Yankee and Surry. It was also addressed in IE Bulletin 79-07 which was sent to all power reactor licensees.

The other issue involves the accuracy of the information input for seismic analyses. In this regard, several potentially unconservative factors were discovered and subsequently addressed in IE Bulletin 79-02 (pipe supports) and 79-04 (valve weights). During resolution of these concerns, inspection by IE and by licensees of the as-built configuration of several piping systems revealed a number of nonconformances to design documents which could potentially affect the validity of seismic analyses. Nonconformances are identified in Appendix A to this bulletin. Because apparently significant nonconformances to design documents have occurred in a number of plants, this issue is generic.

The staff has determined, where design specifications and drawings are used to obtain input information for seismic analysis of safety-related piping systems, that it is essential for these documents to reflect as-built configurations. Where subsequent use, damage or modifications affect the condition or configuration of safety-related piping systems as described in documents from which seismic analysis input information was obtained, the licensee must consider the need to re-evaluate the seismic analyses to consider the as-built configuration.

Action to be taken by Licensees and Permit Holders:

All power reactor facility licensees and construction permit holders are requested to verify, unless verified to an equivalent degree within the last 12 months, that the seismic analysis applies to the actual configuration of safetyrelated piping systems. The safety related piping includes Seismic Category I systems as defined by Regulatory Guide 1.29, "Seismic Design Classification" Revision 1, dated August 1, 1973 or as defined in the applicable FSAR. For older plants, where Seismic Category I requirements did not exist at the time of licensing, it must be shown that the actual configuration of these safety-related systems meets design requirements.

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Specifically, each licensee is requested to:

1. Identify inspection elements to be used in verifying that the seismic analysis input information conforms to the actual configuration of safety-related systems. For each safety-related system, submit a list of design documents, including title, identification number, revision, and date, which were sources of input information for the seismic analyses. Also submit a description of the seismic analysis input information which is contained in each document. Identify systems or portions of systems which are planned to be inspected during each sequential inspection identified in Items 2 and 3. Submit all of this information within 30 days of the date of this bulletin.

2. For portions of systems which are normally accessible*, inspect one system in each set of redundant systems and all nonredundant systems for conformance to the seismic analysis input information set forth in design documents. Include in the inspection: pipe run geometry; support and restraint design, locations, function and clearance (including floor and wall penetration); embedments (excluding those covered in IE Bulletin 79-02); pipe attachements; and valve and valve operator locations and weights (excluding those covered in IE Bulletin 79-04). Within 60 days of the date of this bulletin, submit a description of the results of this inspection. Where nonconformances are found which affect operability of any system, the licensee will expedite completion of the inspection described in Item 3.

3. In accordance with Item 2, inspect all other normally accessible safety-related systems and all normally inaccessible safety-related systems. Within 120 days of the date of this bulletin, submit a description of the results of this inspection.

4. If nonconformances are identified:

   A. Evaluate the effect of the nonconformance upon system operability under specified earthquake loadings and comply with applicable action statements in your technical specifications including prompt reporting.

   B. Submit an evaluation of identified nonconformances on the validity of piping and support analyses as described in the Final Safety Analysis Report (FSAR) or other NRC approved documents. Where you determine that reanalysis is necessary, submit your schedule for: (i) completing the reanalysis, (ii) comparisons of the results to FSAR or other NRC approved acceptance criteria and (iii) submitting descriptions of the results of reanalysis.

*Normally accessible refers to those areas of the plant which can be entered during reactor operation.

C. In lieu of B, submit a schedule for correcting nonconforming systems so that they conform to the design documents. Also submit a description of the work required to establish conformance.

D. Revise documents to reflect the as-built conditions in plant, and describe measures which are in effect which provide assurance that future modifications of piping systems, including their supports, will be reflected in a timely manner in design documents and the seismic analysis.

Facilities holding a construction permit shall inspect safety-related systems in accordance with Items 2 and 3 and report the results within 120 days.

Reports shall be submitted to the Regional Director with copies to the Director of the Office of Inspection and Enforcement and the Director of the Division of Operating Reactors, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

Approved by GAO (R0072); clearance expires 7/31/80. Approval was given under a blanket clearance specifically for generic problems.

LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS

Bulletin No.	Subject	Date Issued	Issued To
	System Piping		OL for action. All BWRs with a CP for information.
79-02 (Rev. 1)	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts	6/21/79	All Power Reactor Facilities with an OL or a CP
79-12	Short Period Scrams at BWR Facilities	5/31/79	All GE BWR Facilities with an OL
79-11	Faulty Overcurrent Trip Device in Circuit Breakers for Engineered Safety Systems	5/22/79	All Power Reactor Facilities with an OL or a CP
79-10	Requalification Training Program Statistics	5/11/79	All Power Reactor Facilities with an OL
79-09	Failures of GE Type AK-2 Circuit Breaker in Safety Related Systems	4/17/79	All Power Reactor Facilities with an OL or CP
79-08	Events Relevant to BWR Reactors Identified During Three Mile Island Incident	4/14/79	All BWR Power Reactor Facilities with an OL
79-07	Seismic Stress Analysis of Safety-Related Piping	4/14/79	All Power Reactor Facilities with an OL or CP
79-06B	Review of Operational Errors and System Mis-alignments Identified During the Three Mile Island Incident	4/14/79	All Combustion Engineer-ing Designed Pressurized Water Power Reactor Facilities with an Operating Licensee
79-06A (Rev 1)	Review of Operational Errors and System Mis-alignments Identified During the Three Mile Island Incident	4/18/79	All Pressurized Water Power Reactor Facilities of Westinghouse Design with an OL

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Bulletin No.	Subject	Date Issued	Issued To
79-06A	Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident	4/14/79	All Pressurized Water Power Reactor Facilities of Westinghouse Design with an OL
79-06	Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident	4/11/79	All Pressurized Water Power Reactors with an OL except B&W facilities
79-05A	Nuclear Incident at Three Mile Island	4/5/79	All B&W Power Reactor Facilities with an OL
79-05	Nuclear Incident at Three Mile Island	4/2/79	All Power Reactor Facilities with an OL and CP
79-04	Incorrect Weights for Swing Check Valves Manufactured by Velan Engineering Corporation	3/30/79	All Power Reactor Facilities with an OL or CP
78-12B	Atypical Weld Material in Reactor Pressure Vessel Welds	3/19/79	All Power Reactor Facilities with an OL or CP
79-03	Longitudinal Welds Defects In ASME SA-312 Type 304 Stainless Steel Pipe Spools Manufactured by Youngstown Welding and Engineering Co.	3/12/79	All Power Reactor Facilities with an OL or CP
79-02	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts	3/2/70	All Power Reactor Facilities with an OL or CP
79-01A	Environmental Qualification of Class 1E Equipment (Deficiencies in the Environmental Qualification of ASCO Solenoid Valves)	6/6/79	All Power Reactor Facilities with an OL or CP
79-01	Environmental Qualification of Class IE Equipment	2/8/79	All Power Reactor Facilities with an OL or CP

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Bulletin No.	Subject	Date Issued	Issued To
78-14	Deterioration of Buna-N Component In ASCO Solenoids	12/19/78	All GE BWR facilities with and OL or CP
78-13	Failures in Source Heads of Kay-Ray, Inc., Gauges Models 7050, 7050B, 7051, 7051B, 7060, 7060B, 7061 and 7061B	10/27/78	All general and specific licensees with the subject Kay-Ray, Inc. gauges
78-12A	Atypical Weld Material in Reactor Pressure Vessel Welds	11/24/78	All Power Reactor Facilities with an OL or CP
78-12	Atypical Weld Material in Reactor Pressure Vessel Welds	9/29/78	All Power Reactor Facilities with an OL or CP
78-11	Examination of Mark I Containment Torus Welds	7/21/78	BWR Power Reactor Facilities for action: Peach Bottom 2 and 3, Quad Cities 1 and 2, Hatch 1, Monticello and Vermont Yankee

APPENDIX A

PLANTS WITH SIGNIFICANT DIFFERENCES BETWEEN ORIGINAL DESIGN AND AS-BUILT CONDITION OF PIPING SYSTEMS

Plant	Difference	Remarks
Surry 1	Mislocated supports. Wrong Support Type. Different Pipe Run Geometry.	As built condition caused majority of pipe overstress problems, not algebraic summation.
Beaver Valley	Not specifically identified. Licensee reported "as-built conditions differ significantly from orginal design."	As built condition resulted in both pipe and support overstress.
Fitzpatrick	IE inspection identified differences similar to Surry.	Licensee is using as built configuration for reanalysis.
Pilgrim	Snubber sizing wrong. Snubber pipe attachment welds and snubber support assembly nonconformances.	Plant shutdown to restore original design condition.
Brunswick 1 and 2	Pipe supports undersize.	Both units shutdown to restore original design condition.
Ginna	Pipe supports not built	Supports were repaired
St. Lucie	Missing seismic supports. Supports on wrong piping	Install corrected supports before start up from refueling.
Nine Mile Point	Missing seismic supports.	Installed supports before startup from refueling.
Indian Point 3	Support location and support construction deviations.	Licensee performing as built verification to be completed by July 1.
Davis-Besse	Gussets missing from main Steam Line Supports.	Supports would be overstressed. Repairs will be completed prior to startup.