UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

April 14, 1979

IE Bulletin No. 79-07

SEISMIC STRESS ANALYSIS OF SAFETY-RELATED PIPING

Description of Circumstances:

In the course of evaluation of certain piping designs, significant discrepancies were observed between the original piping analysis computer code used to analyze earthquake loads and a currently acceptable computer code developed for this purpose. This problem resulted in the Nuclear Regulatory Commission order to shutdown five power reactors whose design had involved the use of the suspect computer codes (IE Information Notice No. 79-06). The difference in predicted piping stresses between the two computer codes is attributable to the fact that the piping analysis code used for a number of piping systems uses an algebraic summation of the loads predicted separately by the computer code for both the horizontal components and for the vertical component of seismic events. This is an incorrect treatment of such loads and was not recognized as such at the time the original analyses were performed. Such codirectional loads should not be algebraically added (with predicted loads in the negative direction offsetting predicted loads in the positive direction) unless certain more complex time-history analyses are performed. Rather, to properly account for the effects of earthquakes on systems important to safety, as required by "Design Bases for Protection Against Natural Phenomena," General Design Criterion 2 of Appendix A to 10 CFR Part 50, such loads should be combined absolutely or, as is the case in the newer codes, using techniques such as the square root of the sum of the squares. These combinations of loads conform to current industry practice.

The inappropriate analytical treatment of load combinations discussed above becomes significant for piping runs in which the horizontal seismic excitation can have both horizontal and vertical components of response on piping systems, and the vertical seismic excitation also has both horizontal and vertical components of response. It is in these runs that the predicted earthquake loads may differ significantly.

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Although the greatest differences in predicted loads would tend to be limited to localized stresses in pipe supports and restraints or in weld attachments to pipes, there could be a substantial number of areas of high stress in piping, as well as a number of areas in which there is potential for damage to adjacent restraints or supports. Any of these situations could have significant adverse effects on the ability of the piping system to withstand seismic events.

The NRC staff has not yet determined that all of the piping systems important to safety that were designed using a piping analysis computer code which contains the algebraic summation error, have been identified. Certain information is needed in order to make this determination.

Action To Be Taken By All Licensees and Permit Holders:

For all power reactor facilities with an operating license or a construction permit:

(1) Identify which, if any, of the methods specified below were employed or were used in computer codes for the seismic analysis of safety related piping in your plant and provide a list of safety systems (or portions thereof) affected:

Response Spectrum Model Analysis:

a. Algebraic (considering signs) summation of the codirectional spatial components (i.e., algebraic summation of the maximum values of the codirectional responses caused by each of the components of earthquake motion at a particular point in the mathematical model).

b. Algebraic (considering signs) summation of the codirectional inter model responses (i.e., for the number of modes considered, the maximum values of response for each mode summed algebraically).

Time History Analysis:

a. Algebraic summation of the codirectional maximum responses or the time dependent responses due to each of the components of earthquake motion acting simultaneously when the earthquake directional motions are not statistically independent.

(2) Provide complete computer program listings for the dynamic response analysis portions for the codes which employed the techniques identified in Item 1 above.

(3) Verify that all piping computer programs were checked against either piping benchmark problems or compared to other piping computer programs. You are requested to identify the benchmark problems and/or the computer programs that were used for such verifications or describe in detail how it was determined that these programs yielded appropriate results (i.e., gave results which corresponded to the correct performance of their intended methodology).

(4) If any of the methods listed in item 1 are identified, submit a plan of action and an estimated schedule for the re-evaluation of the safety related piping, supports, and equipment affected by these analysis techniques. Also provide an estimate of the degree to which the capability of the plant to safely withstand a seismic event in the interim is impacted.

The responses for Items 1, 2 and 3 above, should include all subsequent piping system additions and modifications. Any re-evaluation required, in conformance with Item 4, should incorporate the "as built" conditions.

Licensees of all operating power reactor facilities should submit the information identified in Items 1 through 4, above, within 10 days of the date of this letter. Holders of construction permits for power reactor facilities should submit this information within 45 days of the date of this letter.

Reports should be submitted to the Director of the appropriate NRC Regional Office and a copy should be forwarded to the NRC Office of Inspection and Enforcement, Division of Reactor Operations Inspection, Washington, D.C., 20555.

Approved by GAO, B180225 (R0072); clearance expires 7-31-80. Approval was given under a blanket clearance specifically for identified generic problems.

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LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS

Bulletin No.	Subject	Date Issued	Issued To
78-05	Malfunctioning of Circuit Breaker Auxiliary Contact Mechanism-General Model CR105X	4/14/78	All Power Reactor Facilities with an OL or CP
78-06	Defective Cutler-Hammer, Type M Relays With DC Coils	5/31/78	All Power Reactor Facilities with an OL or CP
78-07	Protection afforded by Air-Line Respirators and Supplied-Air Hoods	6/12/78	All Power Reactor Facilities with an OL, all class E and F Research Reactors with an OL, all Fuel Cycle Facilities with an OL, and all Priority 1 Material Licensees
78-08	Radiation Levels from Fuel Element Transfer Tubes	6/12/78	All Power and Research Reactor Facilities with a Fuel Element transfer tube and an OL.
78-09	BWR Drywell Leakage Paths Associated with Inadequate Drywell Closures	6/14/79	All BWR Power Reactor Facilities with an OL or CP
78-10	Bergen-Paterson Hydraulic Shock Suppressor Accumulator Spring Coils	6/27/78	All BWR Power Reactor Facilities with an OL or CP

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Bulletin No.	Subject	Date Issued	Issued To
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
78-12	Atypical Weld Material in Reactor Pressure Vessel Welds	9/29/78	All Power Reactor Facilities with an OL or CP
78-12A	Atypical Weld Material in Reactor Pressure Vessel Welds	11/24/78	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
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
78-14	Deterioration of Buna-N Components In ASCO Solenoids	12/19/78	All GE BWR 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
79-02	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts	3/2/79	All Power Reactor Facilities with an OL or CP
79-03	Longitudinal Weld 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
79-04	Incorrect Weights for Swing Check Valves Manufactured by Velan Engineering Corporation	3/30/79	All Power Reactor Facilities with an OL or CP
79-05	Nuclear Incident at Three Mile Island	4/1/79	All B&W Power Reactor Facilities with an OL
79-05A	Nuclear Incident at	4/5/79	All B&W Power Reactor Facilities 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 Reactor Facilities Except B&W Facilities
79-06-A	Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident	4/11/79	All Westinghouse PWR Facilities with an OL
79-06-B	Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident	4/11/79	All Combustion Engineering PWR Facilities with an OL