Category 4
#4

PUBLIC SERVICE COMPANY OF COLORADO
P. O. BOX 840 · DENVER, COLORADO 80201

C. K. MILLEN
SENIOR VICE PRESIDENT

December 15, 1976
Fort St. Vrain
Unit No. 1
P-76276

Mr. R. P. Denise
Asst. Dir. for Advanced Reactors
Nuclear Regulatory Commission
Division of Reactor Licensing
7920 Norfolk
Bethesda, MD 20034

Docket No. 50-267

Gentlemen:

We submitted, for your consideration, a request for a temporary change to Technical Specification LCO 4.2.2.d), on December 8, 1976. This request was subsequently granted by the Commission and consisted of a seven day period of relief from the conditions of the referenced portion of the Technical Specification.

Maintenance work on the subject pump was completed on Saturday, December 11, and the pump was installed and testing began December 12. Testing has indicated that the pump performance is only marginally acceptable and at the present reactor primary system pressure of 600#, it has the capacity to supply makeup water to only two helium circulators. Internal inspection of the pump has revealed that the impellers are badly worn and require replacement. The supplier of the pump informs us it will be necessary to cast replacement impellers. Casting and machining will take approximately 8 to 9 days. Upon receipt of these new parts it will take approximately two days to reassemble the pump and another two days to reinstall and test it.

We are, therefore, requesting a 14 day extension for relief from the provisions of LCO 4.2.2.d). We also request that we be allowed to raise the power limitation from the 28% limit included in the 12/8/76 request to 35% of rated reactor power to allow us to proceed with our rise to power testing.

Your earliest response to this request is appreciated.

Very truly yours,

C. K. Millen
Senior Vice President

CKM:il

Public Service Company of Colorado
P. O. Box 361, Platteville, CO 80651

Category 4
#5

November 17, 1977
Fort St. Vrain
Unit No. 1
P-77229

Mr. Richard P. Denise
Asst. Director for Special Projects
Division of Project Management
Nuclear Regulatory Commission
Washington, DC 20555

Docket No. 50-267

Gentlemen:

Subject: Supports for 2" and Under Piping
Ref: P-77199, 9-29-77

In correspondence P-77199, dated 9/29/77, I informed you of a deficiency identified in the supports for 2" and under piping classified as Seismic Class I. I also indicated that the evaluation and corrective action would be completed in eight to ten weeks, during the period 12-1-77 to 12-15-77.

Based on our findings to date, the following revised schedule is submitted for your information:

1.	Complete walkdown of 1200 Isometric Drawings not previously reviewed.	11-18-77
2.	Complete physical work identified by walkdown in 1) above.	11-30-77
3.	Complete re-walkdown of 400 Isometric Drawings previously reviewed.	12-15-77
4.	Complete physical work identified by walkdown in 3) above.	12-31-77
5.	Complete documentation update with all corrections identified in 1) and 3) above.	3-15-77

If you have any questions, please let me know.

Very truly yours,

J K Fuller

J. K. Fuller
Vice President Engineering and Planning

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Category 14
#2

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

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.

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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 safety-related 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. The action items that follow apply to all safety-related piping 2-1/2-inches in diameter and greater and to seismic Category I piping, regardless of size which was dynamically analyzed by computer. 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, utilizing piping 2-1/2 inches in diameter and greater, meets design requirements.

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 attachments 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.

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

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.

   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.

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.	Plan 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 to original design.	Supports were repaired during refueling outage.
St. Lucie	Missing seismic supports. Supports on wrong piping.	Install corrected supports before start up from refueling.

APPENDIX A

Plant	Difference	Remarks
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 over-stressed. Repairs will be completed prior to start-up.

LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS

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	9/26/78	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
78-12A	Atypical Weld Material in Reactor Pressure Vessel Welds	11/24/78	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
78-12B	Atypical Weld Material in Reactor Pressure Vessel Welds	3/19/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (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 Kay-Ray Gauges

78-14	Deterioration of Buna-N Components in ASCO Solenoids	12/19/78	All GE BWR facilities with an Operating License (OL) or Construction Permit (CP)
79-01	Environmental Qualification of Class IE Equipment	2/8/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
79-01A	Environmental Qualification of Class IE Equipment	6/6/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
79-02	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts	3/8/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
79-02 (Rev. 1)	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts	6/21/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
79-03	Longitudinal Weld Defects In ASME SA-312 Type 304 Stainless Steel Pipe Spools Manufactured by Youngstown Welding and Engineering Company	3/12/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
79-04	Incorrect Weights for Swing Check Valves Manufactured by Velan Engineering Corporation	3/30/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)

79-05	Nuclear Incident at Three Mile Island	4/1/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
79-05A	Nuclear Incident at Three Mile Island	4/5/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
79-05B	Nuclear Incident at Three Mile Island	4/21/79	All B&W Power Reactor Facilities with an Operating License (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
79-06A	Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident	4/14/79	All Westinghouse PWR Facilities with an Operating License (OL)
79-06A (Rev. 1)	Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident	4/18/79	All Pressurized Water Power Reactor Facilities of Westinghouse Design with an Operating License (OL)
79-06B	Review of Operational Errors and System Misalignments Identified During The Three Mile Island	4/14/79	All Combustion Engineering PWR Facilities with an Operating License (OL)
79-07	Seismic Stress Analysis of Safety-Related Piping	4/14/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)

79-08	Events Relevant to BWR Reactors Identified During Three Mile Island Incident	4/14/79	All BWR Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
79-09	Failures of GE Type AK-2 Circuit Breaker in Safety Related Systems	5/11/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
79-10	Requalification Training Program Statistics	5/11/79	All Power Reactor Facilities with an Operating License (OL)
79-11	Faulty Overcurrent Trip Device in Circuit Breakers for Engineered Safety Systems	5/22/79	All Power Reactor Facilities with an Operating License (OL) or a Construction Permit (CP)
79-12	Short Period Scrams at BWR Facilities	5/31/79	All Power Reactor Facilities with an Operating License (OL) or a Construction Permit (CP)
79-13	Cracking in Feedwater System Piping	6/25/79	All PWR with an Operating License (OL) for action. All BWR with a Construction Permit (CP) for information

CENTRAL FILES
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 1000
ARLINGTON, TEXAS 76012

Category 14
#3

July 18, 1979

Docket No. 50-267

Public Service Company of Colorado
ATTN: Mr. C. K. Millen
Senior Vice President
P. O. Box 840
Denver, Colorado 80201

Gentlemen:

IE Bulletin 79-14 is revised to limit the scope of work required. The changes are indicated on the enclosed replacement page for the bulletin. If you desire additional information regarding this matter, please contact this office.

Sincerely,

James H Sniezek /acting

Karl V. Seyfrit
Director

Enclosure:
IE Bulletin No. 79-14
Revision 1

cc: D. W. Warembourg, Nuclear Production Manager
Fort St. Vrain Nuclear Station
P. O. Box 368
Platteville, Colorado 80651

L. Brey, Manager, Quality Assurance

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Category 14
#6

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

Supplement IE Bulletin No. 79-14
Date: August 15, 1979

SEISMIC ANALYSIS FOR AS-BUILT SAFETY-RELATED PIPING SYSTEMS

Description of Circumstances:

IE Bulletin No. 79-14 was issued on July 2, 1979, and revised on July 18, 1979. The Bulletin requested licensees to take certain actions to verify that seismic analyses are applicable to as-built plants. This supplement to the Bulletin provides additional guidance and definition of Action Items 2, 3, and 4.

To comply with the requests in IE Bulletin 79-14, it will be necessary for licensees to do the following:

2. Inspect Part of the Accessible Piping

   For each system selected by the licensee in accordance with Item 2 of the Bulletin, the licensee is expected to verify by physical inspection, to the extent practicable, that the inspection elements meet the acceptance criteria. In performing these inspections, the licensee is expected to use measuring techniques of sufficient accuracy to demonstrate that acceptance criteria are met. Where inspection elements important to the seismic analysis cannot be viewed because of thermal insulation or location of the piping, the licensee is expected to remove thermal insulation or provide access. Where physical inspection is not practicable, e.g., for valve weights and materials of construction, the license is expected to verify conformance by inspection of quality assurance records. If a nonconformance is found, the licensee is expected in accordance with Item 4 of the Bulletin to perform an evaluation of the significance of the nonconformance as rapidly as possible to determine whether or not the operability of the system might be jeopardized during a safe shutdown earthquake as defined in the Regulations. This evaluation is expected to be done in two phases involving an initial engineering judgement (within 2 days), followed by an analytical engineering evaluation (within 30 days). Where either phase of the evaluation shows that system operability is in jeopardy, the licensee is expected to meet the applicable technical specification action statement and complete the inspections required by Item 2 and 3 of the Bulletin as soon as possible. The licensee must report the results of these inspections in accordance with the requirements for content and schedule as given in Item 2 and 3 of the Bulletin.

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3. Inspect Remaining Piping

The licensee is expected to inspect, as in Item 2 above, the remaining safety-related piping systems which were seismically analyzed and to report the results in accordance with the requirements for content and schedule as given in Item 3 of the Bulletin.

4A. Evaluate Noncomformances

With regard to Item 3A for the Bulletin, the licensee is expected to include in the initial engineering judgement his justification for continued reactor operation. For the analytical engineering evaluation, the licensee is expected to perform the evaluation by using the same analytical technique used in the seismic analysis or by an alternate, less complex technique provided that the licensee can show that it is conservative.

If either part of the evaluation shows that the system may not perform its intended function during a design basic earthquake, the licensee must promptly comply with applicable action statements and reporting requirements in the Technical Specifications.

4B. Submit Nonconformance Evaluations

The licensee is expected to submit evaluations of all nonconformances and, where the licensee concludes that the seismic analysis may not be conservative, submit schedules for reanalysis in accordance with Item 4B of the Bulletin or correct the noncomformances.

4C. Correct Nonconformances

If the licensee elects to correct nonconformances, the licensee is expected to submit schedules and work descriptions in accordance with Item 4C of the Bulletin.

4D. Improve Qualtiy Assurance

If noncomformances are identified, the licensee is expected to evaluate and improve quality assurance procedures to assure that future modifications are handled efficiently. In accordance with Item 4D of the Bulletin, the licensee is expected to revise design documents and seismic analyses in a timely manner.

The schedule for the action and reporting requirements given in the Bulletin as originally issued remains unchanged.

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 SIX MONTHS

Bulletin No.	Subject	Date Issued	Issued To
79-01A	Environmental Qualification of Class IE Equipment	6/6/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
79-02	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts	3/8/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
79-02 (Rev. 1)	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts	6/21/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
79-03	Longitudinal Weld Defects In ASME SA-312 Type 304 Stainless Steel Pipe Spools Manufactured by Youngstown Welding and Engineering Company	3/12/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
79-04	Incorrect Weights for Swing Check Valves Manufactured by Velan Engineering Corporation	3/30/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
79-05	Nuclear Incident at Three Mile Island	4/1/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)

79-05A	Nuclear Incident at Three Mile Island	4/5/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
79-05B	Nuclear Incident at Three Mile Island	4/21/79	All B&W Power Reactor Facilities with an Operating License (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
79-06A	Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident	4/14/79	All Westinghouse PWR Facilities with an Operating License (OL)
79-06A (Rev. 1)	Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident	4/18/79	All Pressurized Water Power Reactor Facilities of Westinghouse Design with an Operating License (OL)
79-06B	Review of Operational Errors and System Misalignments Identified During The Three Mile Island	4/14/79	All Combustion Engineering PWR Facilities with an Operating License (OL)
79-07	Seismic Stress Analysis of Safety-Related Piping	4/14/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)

79-08	Events Relevant to BWR Reactors Identified During Three Mile Island Incident	4/14/79	All BWR Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
79-09	Failures of GE Type AK-2 Circuit Breaker in Safety Related Systems	5/11/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
79-10	Requalification Training Program Statistics	5/11/79	All Power Reactor Facilities with an Operating License (OL)
79-11	Faulty Overcurrent Trip Device in Circuit Breakers for Engineered Safety Systems	5/22/79	All Power Reactor Facilities with an Operating License (OL) or a Construction Permit (CP)
79-12	Short Period Scrams at BWR Facilities	5/31/79	All Power Reactor Facilities with an Operating License (OL) or a Construction Permit (CP)
79-13	Cracking In Feedwater System Piping	6/25/79	All PWRs with an Operating License (OL) for action. All BWR with a Construction Permit (CP) for information
79-14	Seismic Analyses for As-Built Safety-Related Piping System	7/2/79	All Power Reactor facilities with an Operting License (OL) or a Construction Permit (CP)
79-15	Deep Draft Pump Deficiencies	7/11/79	All Power Reactor Facilities with a Construction Permit and/or Operating License (OL)
79-16	Vital Area Access Controls	7/26/79	All Power Reactors with an Operating License (OL) or anticipating fuel loading prior to January 1981.
79-17	Pipe Cracks in Stagnant Borated Water Systems at PWR Plants	7/26/79	All PWRs with Operating License (OL)

79-18	Audibility Problems Encountered on Evacuation of Personnel from High-Noise Areas	8/6/79	All Power Reactor Facilities with an Operating License (OL)
79-19	Packaging Low-Level Radioactive Waste for Transport and Burial	8/10/79	All Power and Research Reactors with Operating Licenses (OLs), fuel facilities except uranium mills, and certain materials licensees
79-20	Packaging Low-Level Radioactive Waste for Transport and Burial	8/10/79	Materials Licensees who did not receive Bulletin No. 79-19
79-21	Temperature Effects on Level Measurements	8/13/79	All PWR's with an Operating License (OL) for action. All BWR's with a Construction Permit (CP) for information

Category 14
#7

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

SEISMIC ANALYSIS FOR AS-BUILT SAFETY-RELATED PIPING SYSTEMS

Description of Circumstances:

IE Bulletin No. 79-14 was issued on July 2, revised on July 18, and first supplemented on August 15, 1979. The bulletin requested licensees to take certain actions to verify that seismic analyses are applicable to as-built plants. Supplement 2 provides the following additional guidance with regard to implementation of the bulletin requirements:

Nonconformances

One way of satisfying the requirements of the bulletin is to inspect safety-related piping systems against the specific revisions of drawings which were used as input to the seismic analysis. Some architect-engineers (A-E) however, are recommending that their customers inspect these systems against the latest revisions of the drawings and mark them as necessary to define the as-built configuration of the systems. These drawings are then returned to the A-E's offices for comparison by the analyst to the seismic analysis input. For licensees taking this approach, the seismic analyst will be the person who will identify nonconformances.

The first supplement to the bulletin provided guidance with regard to evaluation of nonconformances. That guidance is appropriate for licensees inspecting against later drawings. The licensee should assure that he is promptly notified when the A-E identifies a nonconformance, that the initial engineering judgment is completed in two days, and that the analytical engineering evaluation is completed in 30 days. If either the engineering judgement or the analytical engineering evaluation indicates that system operability is in jeopardy, the licensee is expected to meet the applicable technical specification action statement.

Visual Approximations

Some licensees are visually estimating pipe lengths and other inspection elements, and have not documented which data have been obtained in that way. Visual estimation of dimensions is not encouraged for most measurements; however, where visual estimates are used, the accuracy of estimation must be within tolerance requirements. Further, in documenting the data, the licensee must specifically identify those data that were visually estimated.

Thermal Insulation

In many areas, thermal insulation interferes with inspection of pipe support details, i.e., attachment welds, saddles, support configuration, etc. In some areas, the presence of thermal insulation may result in unacceptably large uncertainties for determination of the location of pipe supports.

Where thermal insulation obstructs inspection of support details, the insulation should be removed for inspection of a minimum of 10% of the obstructed pipe supports in both Items 2 and 3 inspections. In the Item 3 response, the licensee should include a schedule for inspecting the remaining supports.

Where necessary to determine the location of pipe supports to an accuracy within design tolerances, thermal insulation must be removed.

Clearances

For exposed attachments and penetrations, licensees are expected to measure or estimate clearances between piping and supports, integral piping attachments (e.g., lugs and gussets) and supports, and piping and penetrations. Licensees are not expected to do any disassembly to measure clearances.

Loose Bolts

Loose anchor bolts are not covered by this bulletin, but are covered by IE Bulletin No. 79-02. Any loose anchor bolts identified during actions taken for this bulletin should be dispositioned under the requirements of IE Bulletin No. 79-02.

Other loose bolts are to be treated as nonconformances if they invalidate the seismic analysis; however, torquing of bolts is not required.

Difficult Access

Areas where inspections are required by the IE Bulletin, but are considered impractical even with the reactor shutdown, should be addressed on a case-by-case basis. Information concerning the burden of performing the inspection and the safety consequence of not performing the inspection should be documented by the licensee and forwarded for staff review.

Schedule

The schedule for the action and reporting requirements given in the IE Bulletin as originally issued remains unchanged.

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

IE Bulletin No. 79-14
Supplement 2
September 7, 1979

LISTING OF IE BULLETINS
ISSUED IN LAST SIX MONTHS

Bulletin No.	Subject	Date Issued	Issued To
79-01A	Environmental Qualification of Class IE Equipment	6/6/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
79-02	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts	3/8/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
79-02 (Rev. 1)	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts	6/21/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
79-03	Longitudinal Weld Defects In ASME SA-312 Type 304 Stainless Steel Pipe Spools Manufactured by Youngstown Welding and Engineering Company	3/12/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
79-04	Incorrect Weights for Swing Check Valves Manufactured by Velan Engineering Corporation	3/30/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
79-05	Nuclear Incident at Three Mile Island	4/1/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)

79-05A	Nuclear Incident at Three Mile Island	4/5/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
79-05B	Nuclear Incident at Three Mile Island	4/21/79	All B&W Power Reactor Facilities with an Operating License (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
79-06A	Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident	4/14/79	All Westinghouse PWR Facilities with an Operating License (OL)
79-06A (Rev. 1)	Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident	4/18/79	All Pressurized Water Power Reactor Facilities of Westinghouse Design with an Operating License (OL)
79-06B	Review of Operational Errors and System Misalignments Identified During The Three Mile Island	4/14/79	All Combustion Engineering PWR Facilities with an Operating License (OL)
79-07	Seismic Stress Analysis of Safety-Related Piping	4/14/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)

79-08	Events Relevant to BWR Reactors Identified During Three Mile Island Incident	4/14/79	All BWR Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
79-09	Failures of GE Type AK-2 Circuit Breaker in Safety Related Systems	5/11/79	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
79-10	Requalification Training Program Statistics	5/11/79	All Power Reactor Facilities with an Operating License (OL)
79-11	Faulty Overcurrent Trip Device in Circuit Breakers for Engineered Safety Systems	5/22/79	All Power Reactor Facilities with an Operating License (OL) or a Construction Permit (CP)
79-12	Short Period Scrams at BWR Facilities	5/31/79	All Power Reactor Facilities with an Operating License (OL) or a Construction Permit (CP)
79-13	Cracking In Feedwater System Piping	6/25/79	All PWRs with an Operating License (OL) for action. All BWR with a Construction Permit (CP) for information
79-14	Seismic Analyses for As-Built Safety-Related Piping System	7/2/79	All Power Reactor facilities with an Operting License (OL) or a Construction Permit (CP)
79-15	Deep Draft Pump Deficiencies	7/11/79	All Power Reactor Facilities with a Construction Permit and/or Operating License (OL)
79-16	Vital Area Access Controls	7/26/79	All Power Reactors with an Operating License (OL) or anticipating fuel loading prior to January 1981.
79-17	Pipe Cracks in Stagnant Borated Water Systems at PWR Plants	7/26/79	All PWRs with Operating License (OL) or a Construction Permit (CP)

79-18	Audibility Problems Encountered on Evacuation of Personnel from High-Noise Areas	8/6/79	All Power Reactor Facilities with an Operating License (OL) or a Construction Permit (CP)
79-19	Packaging Low-Level Radioactive Waste for Transport and Burial	8/10/79	All Power and Research Reactors with Operating Licenses (OLs), fuel facilities except uranium mills, and certain materials licensees
79-20	Packaging Low-Level Radioactive Waste for Transport and Burial	8/10/79	Materials Licensees who did not receive Bulletin No. 79-19
79-21	Temperature Effects on Level Measurements	8/13/79	All PWR's with an Operating License (OL) or Construction Permit (CP)

Category 14
#8

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October 15, 1979
Fort St. Vrain
Unit No. 1
P-79238

Mr. Karl V. Seyfrit, Director
Nuclear Regulatory Commission
Region IV
Office of Inspection and Enforcement
611 Ryan Plaza Drive
Suite 1000
Arlington, Texas 76012

Subject: Safety Related Piping System Audit

Reference: PSC Letter P-79161 dated Aug. 1, 1979
PSC Letter P-79198 dated Sept.4, 1979

Dear Mr. Seyfrit:

On September 24, 1979, Public Service Company of Colorado notified the NRC Resident Inspector that the inconsistencies noted in the 12 pipe hangers of the random sample audit conducted on the safety related piping systems and reported to the NRC in PSC Letter P-79198 dated September 4, 1979 had been resolved and that the Fort St. Vrain plant would resume power operation effective September 25, 1979.

The resolutions of the discrepancies in the 12 hangers involved in the sample audit were discussed with the NRC Resident Inspector during the week of September 24, 1979. The lines associated with the hangers in question required the utilization of sophisticated analytical techniques to ascertain system operability which could not be completed in the alloted time thereby necessitating a plant shutdown until the anlaysis work could be completed. The lines and hangers have since been analyzed using sophisticated analytical techniques when necessary and, as a result, has allowed PSC Engineering to conclude that there was no system impairment in eleven of the twelve cases. The twelfth case involved a very complex hanger (11A-H-46130) containing numerous Class I and Class II lines that required model simplification techniques to analyze without exceeding computer core capability on the available computers. PSC Engineering continued the analysis to the point where it was concluded that the hanger would meet seismic design requirements if eight fillet welds were increased from 3/16" to 5/16" as originally specified on the hanger drawing. A craft action was written and the additional weldment was applied to the hanger. The analysis required to assess the acceptability of the as-found condition of the hanger was both time consuming and costly. For these reasons,

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the analysis to determine operability was discontinued at the point of craft action completion. At this point in time system operability was arbitrarily assumed to be in jeopardy when, in fact, such might not have been the case.

As a result of the sample audit program PSC concludes: (1) the use of simplified analysis techniques is conservative in that system impairment may be indicated, when in reality, system impairment would not be indicated when analyzed by sophisticated analytical techniques, (2) the simplified techniques would have been satisfied had corrective action been taken to correct the discrepancies to return the hangers to it's originally intended configuration and (3) even though discrepancies exist, system operability is not in jeopardy.

Based on the above, PSC intends to implement the requirement of I & E Bulletin 79-14 as follows:

1. A 100% audit of all accessible 2 1/2" and larger and 2" and under computer analyzed Class I piping. The piping audit is being conducted at this time. The systems or portions of systems required to facilitate reactor primary coolant depressurization and decay heat removal through PCRV liner cooling are being audited on a first priority basis.

2. Discrepancies found during the field audit shall be resolved by expediently returning the discrepant item to it's originally intended configuration. Any discrepant item that can not be resolved in this manner will be evaluated per item 3.

3. An evaluation will be performed based on engineering judgement using the as-built drawings. The evaluation will be completed within two days. If this evaluation indicates that the affected system operability is in jeopardy, the applicable Technical Specification action will be taken.

4. If the initial engineering judgement indicates that the system operability is not in jeopardy, an analytical engineering evaluation will be conducted and completed within 30 days. The technique used in the analytical evaluation will be based upon simplified analytical techniques. If the simplified analytical technique indicates that the piping or hanger does not meet the design requirements of the simplified technique, corrective action to bring the piping or hanger into compliance with the simplified analytical techniques will be completed within the 30 day period. If the piping system or hanger analysis and craft action (if required) is not completed within this 30 day period, the applicable Technical Specification action will be taken. In addition, approximately five per cent of the discrepancies will be analyzed using a sophisticated analytical technique to verify the judgemental evaluations and the conservatism of the simplified analytical techniques. At the completion of the program, all affected documents will be revised to reflect the as-built conditions in the plant. PSC's current

engineering and design procedures will be modified, if required, to provide assurance that modifications to systems include updating of piping supports. The current PSC Engineering procedures are adequate but may be further refined as we proceed through the as-built programs.

All items requiring Technical Specification reporting will be reported in accordance with the applicable directives contained therein. All "discrepancies" will be reported to the Commission on 60-day intervals. The 60 day report shall include a short summary of all discrepancies and corrective actions. PSC intends to correct all discrepancies in a timely manner throughout the duration of the hanger program.

Based upon the scope and available manpower, we anticipate a projected project completion date of December, 1980 with a pessimistic completion date of March, 1981. If there are any questions concerning this program, please contact this office.

Very truly yours,

Frederic E. Swart

Frederic E. Swart
Nuclear Project Manager

FES/LMM/scp