NOTICE OF VIOLATION
AND
PROPOSED IMPOSITION OF CIVIL PENALTY

Consumers Power Company
Palisades Nuclear Generating Plant

Docket No. 50-255
License No. DPR-20
EA 91-125

During two NRC inspections conducted on September 19, 1990, through April 18, 1991, and June 10-21, 1991, violations of NRC requirements were identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (1991), the Nuclear Regulatory Commission proposes to impose a civil penalty pursuant to Section 234 of the Atomic Energy Act of 1954, as amended (Act), 42 U.S.C. 2282, and 10 CFR 2.205. The particular violations and associated civil penalty are set forth below:

I. 10 CFR Part 50, Appendix B, Criterion III, required, in part, that measures be established to assure that regulatory requirements and design bases are correctly translated into design documents. Also, design control measures shall provide for verifying or checking the adequacy of design.

A. The Palisades Nuclear Power Plant Updated Final Safety Analysis Report (UFSAR), Section 5.7.4.1, "Seismic Analysis of CPCo Design Class 1 Piping," states that piping systems were analyzed for each horizontal direction combined simultaneously with the vertical direction (absolute sum method).

Contrary to the above, adequate measures were not established to assure that design bases were correctly translated into design documents. Specifically, Palisades Specification M-195, "Requirements for the Design and Analysis of Palisades Plant Safety Related Piping and Instrument Tubing," Revision 1, dated May 9, 1990, Section 5.10.4.1.2, "Combination of Directional Responses," which implements UFSAR Section 5.7.4.1, specified that when the 1/2% damping curves were used, the vertical and horizontal responses were to be combined using the square root sum of the squares (SRSS) methods. The SRSS method is less conservative than the absolute sum method.

B. The Palisades Nuclear Power Plant, UFSAR, Section 5.7.1.3, "Floor Design Response Spectra," stated that floor response spectra peaks for the containment building natural frequencies were widened ±10% to account for variations in soil and structural material properties.

Contrary to the above, adequate measures were not established to assure that design bases were correctly translated into design documents. Specifically, Palisades Specification M-195, "Requirements for the Design and Analysis of Palisades Plant Safety Related Piping and Instrument Tubing," Revision 1, dated May 9, 1990, Attachment 3, "Original Palisades Plant Response Spectra and Building Displacements," documents that the response spectra peaks for the first natural frequencies of the containment were only widened between 6.97% and 7.77% on five of the seven floor elevations. For the second natural frequency of the containment, the response spectra peaks for four of the seven floor elevations were widened less than 10%. For the third natural frequency of the containment internal structure, the peak was not widened for elevation 649 feet.

C. The Palisades Nuclear Generating Plant, UFSAR, Section 5.7.2.1, "Containment Building," stated that the results of the final seismic dynamic analyses were shown in Figure 5.7-7, "Containment Building Maximum Seismic Response (OBE)," which gave zero period accelerations (ZPA) values for various elevations in containment.

Contrary to the above, adequate measures were not established to assure that design bases were correctly translated into design documents. Specifically, Palisades Specification M-195, "Requirements for the Design and Analysis of Palisades Plant Safety Related Piping and Instrument Tubing," Revision 1, dated May 9, 1990, Attachment 3, "Original Palisades Plant Response Spectra and Building Displacement," specified ZPA values that were less conservative than values listed in the UFSAR. For example, for elevation 590 feet, the ZPA value in UFSAR Figure 5.7-7 is 0.119, and is 0.100 in M-195, Attachment 3.

D. The Palisades Nuclear Power Plant, UFSAR, Section 5.7.4.1, "Seismic Analysis of CPCo Design Class 1 Piping," as implemented by Palisades Specification M-195, "Requirements for the Design and Analysis of Palisades Plant Safety Related Piping and Instrument Tubing", Revision 1, dated May 9, 1990, Paragraph 5.10.4.1, "Seismic Inertia," require that for piping systems spanning two or more elevations, the response spectrum curve for the elevation closest to and higher than the center of mass of the piping system be used.

Contrary to the above, adequate measures were not established to assure that design bases were correctly translated into design documents. Specifically, Calculation No. SGRP-PDS-033, "Pipe Stress Analysis of Steam Generator E50A Main Steam System," Revision 1, dated September 6, 1990, and Revision 2, dated January 21, 1991, Paragraph 3.7, "Applicable Seismic Input," used a response spectrum curve for structural elevation 649 which was 16 feet lower than the center of mass of the piping system.

E. Palisades Specification M-195, "Requirements for the Design and Analysis of Palisades Plant Safety Related Piping and Instrument Tubing," Revision 1, dated May 9, 1990, Paragraph 5.10.4.2, "Seismic Anchor Movements (SAM)," specified that the total seismic displacement will be used in the analysis of branch piping.

Contrary to the above, adequate measures were not established to assure that design bases were correctly translated into design documents. Specifically, Calculation SGRP-PD5-033, "Pipe Stress Analysis of Steam Generator E50A Main Steam System," Revision 1, dated September 6, 1990, used SAM displacements from structural elevation 649 feet which neglected the additional SAM displacement from the actual attachment point of the piping system to the steam generator at elevation 677 feet.

F. Palisades Specification M-195, "Requirements for the Design and Analysis of Palisades Plant Safety Related Piping and Instrument Tubing," Revision 1, dated May 9, 1990, Paragraph 5.10.4.2, "Seismic Anchor Movements (SAM)," specified that individual structure SAM displacements shall be taken from Attachment 4 to M-195 for the Code Case N-411 seismic criteria.

Contrary to the above, adequate measures were not established to assure that design bases were correctly translated into design documents. Specifically, Bechtel Specification No. 20557-G-001P, "Design Criteria Documents for Palisades Nuclear Plant Steam Generator Replacement," Revision 3, dated October 31, 1990, Paragraph 4.4.2.4.2, "Seismic Anchor Movements," did not include the SAM displacements from Attachment 4 to M-195 for the Code Case N-411 seismic criteria.

G. The Palisades Nuclear Power Plant, UFSAR, Section 5.10.1.1, "CPCo Design Class 1 Piping," stated that piping was designed to USA Standard B31.1.0-1967, "Power Piping Code (Code)." Paragraph 120.2.4 of the Code requires that for supplementary steel, no modification for allowable stresses for hydrostatic test periods will be permitted.

Contrary to the above, adequate measures were not established to assure that design bases were correctly translated into design documents. Specifically, Palisades Specification C-173, "Technical Requirements for the Analysis and Design of Safety Related Pipe Supports," Revision 2, dated November 21, 1990, Tables 1.0 and 2.0, specified increased allowables for supplementary steel during hydrostatic test periods.

H. The Palisades Nuclear Power Plant, UFSAR, Section 5.10.1.1, "CPCo Design Class 1 Piping," stated that piping was designed to USA Standard B31.1.0-1967, "Power Piping Code." Paragraph 121.2.1 of the Code specified that fixed pipe restraints be structurally suitable to withstand the thrust, movements and other loads imposed during the [thermal] expansion and contraction of piping.

Contrary to the above, adequate measures were not established to assure that design bases were correctly translated into design documents. Specifically, Palisades Specification C-173, "Technical Requirements for the Analysis and Design of Safety Related Pipe

Supports," Section 5.4.2, "Friction Load," Revision 1, specified that the existing pipe restraints be analyzed for friction forces caused by dead loads only and did not include friction forces caused by the loads due to thermal expansion and contraction on the pipe supports.

I. Palisades Specification C-173, "Technical Requirements for the Analysis and Design of Safety Related Pipe Supports," Paragraph 5.10.3, "Shear Lugs," Revision 1, specifies that when more than half of the lugs were considered effective, the load was to be assigned based on the relative flexibility of the supporting members.

Contrary to the above, adequate measures were not established to assure that design bases were correctly translated into design documents. Specifically, Calculation MSA-PD-EB1-H3, "Pipe Support Design for Main Steam System," Revision 2, dated January 21, 1991, assumed that the restraining forces were equally distributed between the only two lugs (more than half of the lugs) even though the flexibility of the supporting members was different by a factor of two.

J. Palisades Specification C-173, "Technical Requirements for the Analysis and Design of Safety Related Pipe Support," Paragraph 5.7.1, "Deflection General Requirements," Revision 1, specifies that the total deflection of the pipe support shall not exceed 1/16 inch.

Contrary to the above, adequate measures were not established to assure that design bases were correctly translated into design documents. Specifically, Calculation MSA-PD-EB1-H3, "Pipe Support Design for Main Steam System," Revision 2, dated January 21, 1991, failed to recognize that the total deflection of the pipe support exceeded 1/16 inch.

K. Bechtel Specification No. 20557-G-001P, "Design Criteria Documents for Palisades Nuclear Plant Steam Generator Replacement Project," Revision 3, dated October 31, 1990, Paragraph 5.4.17.1.1, "Baseplate Design-General," specified that analyses must account for expansion anchor bolt flexibilities as applicable in Appendix B of the specification.

Contrary to the above, adequate measures were not established to assure that design bases were correctly translated into design documents. Specifically, Calculation MSH-PD-EB1-H3, "Pipe Support Design for Main Steam System," Revision 2, dated January 21, 1991, used a flexibility value derived from expansion anchor data which was not applicable to the four through-bolted one inch diameter rods attaching the baseplate to the structure.

L. Bechtel Specification No. 20557-G-001P, "Design Criteria Documents for Palisades Nuclear Plant Steam Generator Replacement Project," Revision 3, Paragraph 4.4.1.4, "Stress Intensification Factors," specified that piping analysis should use the applicable ANSI B31.1 stress intensification factors. The ANSI B31.1 stress intensification factor (SIF) equation, taken from 1973 Edition with Summer of 1973 Addenda, stated that it was applicable only if certain field installation conditions were met.

Contrary to the above, Calculation SGRP-PDS-003, "Pipe Stress Analysis of Steam Generator E50A Blowdown Piping," Revision 5, dated August 21, 1990, utilized the ANSI B31.1 Code equation to calculate SIFs for several branch connections but did not specify nor verify that the Code specified conditions were met.

M. The Palisades Nuclear Power Plant, UFSAR, Section 5.10.1.2, stated that pipe supports were designed using the criteria of the American Institute of Steel Construction (AISC) Specification, Seventh Edition, 1970. Part 4 of the AISC Specification for prequalified welded joints stated that fillet welds for skewed T-joints were limited to a minimum angle of 60° and that for angles less than 60°, the weld was considered a partial penetration groove weld.

Contrary to the above, adequate measures were not established to assure that design bases were correctly translated into design documents. Specifically, for Drawing No. M101-6010, "Pipe Support Number SGAB-PD-H9," Revision 3, dated November 10, 1990, Field Change Notice No. 293 resulted in a skewed T-joint weld angle of approximately 49° and the affected portion of the weld was not changed from a fillet weld to a partial penetration groove weld.

N. The Palisades Nuclear Power Plant UFSAR, Section 5.7.4, "Seismic Analysis of CPCo Design Class 1 Piping," stated that use of the higher damping values, specified in the American Society of Mechanical Engineers (ASME) Section III, Code Case N-411, required adherence to the conditions specified in Regulatory Guide 1.84, Revision 24. Regulatory Guide 1.84, Revision 24, included the condition that analyses using these damping values had to employ current seismic spectra and procedure. The current Standard Review Plan, NUREG-0800, Revision 2, July 1981, stated that seismic analysis of equipment supported at two or more locations required the use of the upper bound envelope of the spectra at all support attachment points.

Contrary to the above, adequate measures were not established to assure that design bases were correctly translated into design documents. Specifically, Calculation SGRP-PDS-002, "Pipe Stress Analysis of Steam Generator E50B, Recirculation Piping Inside Containment," Revision 8, January 10, 1991, did not use upper bound envelope seismic response spectra values in that it utilized spectra from elevation 649 feet when the highest structural attachment point was on the steam generator at elevation 661 feet.

O. Bechtel Specification No. 20557-G-001P, "Design Criteria for Palisades Nuclear Plant Steam Generator Replacement Project," Revision 3, dated October 31, 1990, Table B-4, as referenced in Paragraph 5.4.17.3.1 of the specification for capacity reduction due to shear cone overlap, stated that, if the spacing was smaller than specified, the allowable anchor bolt design capacity shall be reduced in proportion to the ratio for the spacing provided to the spacing required.

Contrary to the above, adequate measures were not established to assure that design bases were correctly translated into design documents. Specifically, Calculation SGRP-PD-H14, "Pipe Support Design for Steam Generator E50B Blowdown," Revision 2, dated January 31, 1991, failed to evaluate the allowable anchor bolt design capacity when the installed configuration had a spacing smaller than specified.

Also, contrary to the above, Revision 3, dated March 1, 1991, of the above listed calculation, did not reduce the anchor bolt capacity by the ratio of the spacing provided to the spacing required, but instead used a methodology based on "reserved" concrete concept which had no previously established basis.

P. Palisades Administrative Procedure No. 9.11, "Engineering Analysis," Revision 4, dated December 28, 1989, Paragraph 6.4.2.b, "Detailed Technical Reviews," stated that detailed review shall verify the accuracy, completeness, and adequacy of the engineering analysis.

Contrary to the above, adequate measures were not established to assure that design bases were correctly translated into design documents. Specifically, the detailed technical review performed for Calculation EA-SC-90-083-01, "Change K-8 Turbine to Class II (675 psi/650°F)," Revision 2, dated November 27, 1990, did not consider the effects of the additional moments caused by the addition of an eccentric reducer nor the effect on the stress intensification factor for the eccentric reducer which was not defined in the piping design Code.

Q. Palisades Specification C-173, "Technical Requirements for the Analysis and Design of Safety Related Pipe Supports," Revision 1, Paragraph 5.11.5, "Rod Hangers," required that when double rod hangers were used on a vertical riser pipe, the hanger components and supporting structures were to be designed to take the total design load on one side.

Contrary to the above, adequate measures were not established to assure that design bases were correctly translated into design documents. Specifically, Calculation EA-03340-HC12-H1, "Safeguards Room Containment Sump Drains Support Package," Revision 3, dated May 28, 1990, for a double rod hanger on a vertical riser pipe, evaluated the hanger components and supporting structures with half of the total design load on each side.

R. The Palisades Nuclear Power Plant UFSAR, Section 5.10.1.1, "CPCo Design Class 1 Piping," stated that piping was designed to USA Standard B31.1.0-1967, "Power Piping Code." Paragraph 127.4.8(c) of the Code stated that branch connections which abut the outside surface of the run wall shall be attached by means of full penetration welds.

Contrary to the above, adequate measures were not established to assure that design bases were correctly translated into design documents. Specifically, instructions given to the welder on Repair Inspection Checklists for welds No. 1 and No. 10 on Drawing 24804973, dated August 23, 1988, and welds No. 1 and No. 14 on Drawing 24804972, dated August 27, 1988, specified attachment welds for all four branch connection as fillet welds. Fillet welds are not full penetration welds.

II. 10 CFR Part 50, Appendix B, Criterion V, required, in part, that activities affecting quality shall be accomplished in accordance with prescribed instructions and procedures.

A. Palisades Administrative Procedure 3.03, "Corrective Action," Revision 4, October 8, 1988, Paragraph 6.5, "Completion of Corrective Actions," stated that if the corrective action taken differs from the proposed action specified by the Plant Review Committee (PRC), the event report shall be returned to the PRC for concurrence.

Contrary to the above, the corrective actions taken on December 27, 1990, for Event Report No. E-PAL-89-030P, in accordance with the licensee's response to the NRC dated December 18, 1989, differed from the actions specified by the PRC and the event report was not returned to the PRC for concurrence. Specifically, the proposed corrective action specified internal visual verification that four welds were full penetration welds, and the actual corrective action consisted of a documentation review and interviews with welding supervisors.

B. Palisades Administrative Procedure 3.07, "Safety Evaluations," Revision 4, dated January 23, 1990, Paragraph 5.2.4, required that when answering each Safety Review question, the preparer list in the safety evaluation FSAR sections affected by the item under review.

Contrary to the above, in Safety Review, PS&L Log No. 90-0797, "Main Steam System," FC-911, Revision 0, dated September 28, 1990, the preparer did not list UFSAR Section 5.7.4, "Seismic Analysis of CPCo Design Class 1 Piping," and consequently failed to note that UFSAR Section 5.7.4.1 and Figure 5.7-27, were directly affected by this change to the facility.

III. 10 CFR Part 50, Appendix B, Criterion XVI, required, in part, that measures be established to assure that nonconformances were promptly identified and corrected.

A. Contrary to the above, the established measures were insufficient to assure that nonconformances were promptly identified and corrected in that the action taken on December 27, 1990, to resolve Event Report E-PAL-89-030P failed to include proper verification of Weld No. 14 on Drawing 24804972 and Weld No. 1 on Drawing 24804973 which were subsequently found to be nonconforming welds. Specifically, the licensee did not verify full weld penetration before closing out the event report.

B. Contrary to the above, during a maintenance outage in May 1990, the licensee identified a leaking weld in the containment spray header, which constituted a nonconformance to the American Society of Mechanical Engineers, Section XI, 1983 Edition, IWA 5250, "Corrective Measures," and failed to assure the nonconformance was promptly corrected. Specifically, the licensee returned the reactor to power with the weld in a nonconforming condition, and did not correct the leaking weld until approximately four months later.

C. Contrary to the above, corrective action taken in response to Palisades Quality Assurance (QA) Audits SGRP-SV-90-A1 and SGRP-SV-90-A2 conducted in February 1990 and July 1990 respectively, did not correct the identified design control program deficiencies in that the same types of design control deficiencies continued to be identified as documented in the Palisades QA Audit SGRP-SV-91-A1 conducted in January and February 1991. Specifically, QA Audit SGRP-SV-91-A1 documented over 100 comments, questions or concerns as examples of failing to meet ANSI N45.2.11 QA requirements for design of nuclear power plants.

IV. 10 CFR 50.59, "Changes, Tests and Experiments," stated that licensees may make changes to the facility as described in the safety analysis report without prior Commission approval unless the proposed change involves an unreviewed safety question, including a reduction in the margin of safety defined in the basis for any technical specification.

Contrary to the above, in the change to the Final Safety Analysis Report (FSAR), dated October 24, 1980, the licensee reduced the margin of safety inherent in the original seismic design basis discussed in Palisades Technical Specification Paragraph 4.16 by increasing the allowable stress value for certain piping from 1.1Sy to 2.4Sh without prior NRC approval and has used this increased stress allowable in all piping analyses since that time.

This is a Severity Level III problem (Supplement I).
Cumulative Civil Penalty - $100,000 (assessed equally among the 24 violations).

Pursuant to the provisions of 10 CFR 2.201, Consumers Power Company (Licensee) is hereby required to submit a written statement or explanation to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, within 30 days of the date of this Notice of Violation and Proposed Imposition of Civil Penalty (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each alleged violation: (1) admission or denial of the alleged violation, (2) the reasons for the violation if admitted, and if denied, the reasons why, (3) the corrective steps that have been taken and the results achieved, (4) the corrective steps that will be taken to avoid further violations, and (5) the date when full compliance will be achieved. If an adequate reply is not received within the time specified in this Notice, an order or a demand for information may be issued as to why the license should not be modified, suspended, or revoked or why such other action as may be proper should not be taken. Consideration may be given to extending the response time for good cause shown. Under the authority of Section 182 of the Act, 42 U.S.C. 2232, this response shall be submitted under oath or affirmation.

Within the same time as provided for the response required above under 10 CFR 2.201, the Licensee may pay the civil penalty by letter addressed to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, with a check, draft, money order, or electronic transfer payable to the Treasurer of the United States in the amount of the civil penalty proposed above, or may protest imposition of the civil penalty in whole or in part, by a written answer addressed to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission. Should the Licensee fail to answer within the time specified, an order imposing the civil penalty will be issued. Should the Licensee elect to file an answer in accordance with 10 CFR 2.205 protesting the civil penalty, in whole or in part, such answer should be clearly marked as an "Answer to a Notice of Violation" and may: (1) deny the violation(s) listed in this Notice, in whole or in part, (2) demonstrate extenuating circumstances, (3) show error in this Notice, or (4) show other reasons why the penalty should not be imposed. In addition to protesting the civil penalty in whole or in part, such answer may request remission or mitigation of the penalty.

In requesting mitigation of the proposed penalty, the factors addressed in Section V.B of 10 CFR Part 2, Appendix C (1991), should be addressed. Any written answer in accordance with 10 CFR 2.205 should be set forth separately from the statement or explanation in reply pursuant to 10 CFR 2.201, but may incorporate parts of the 10 CFR 2.201 reply by specific reference (e.g., citing page and paragraph numbers) to avoid repetition. The attention of the Licensee is directed to the other provisions of 10 CFR 2.205, regarding the procedure for imposing a civil penalty.

Upon failure to pay any civil penalty due which subsequently has been determined in accordance with the applicable provisions of 10 CFR 2.205, this matter may be referred to the Attorney General, and the penalty, unless compromised, remitted, or mitigated, may be collected by civil action pursuant to Section 234c of the Act, 42 U.S.C. 2282c.

The response noted above (Reply to Notice of Violation, letter with payment of civil penalty, and Answer to a Notice of Violation) should be addressed to:

Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region III, 799 Roosevelt Road, Glen Ellyn, Illinois 60137, and a copy to the NRC Resident Inspector at the Palisades Nuclear Generating Plant.

FOR THE NUCLEAR REGULATORY COMMISSION

Carl J Paperiello for

A. Bert Davis
Regional Administrator

Dated at Glen Ellyn, Illinois
this 15th day of January 1992

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RidS

Docket No. 50-255

Consumers Power Company
ATTN: Gerald B. Slade
General Manager
Palisades Nuclear Generating Plant
27780 Blue Star Memorial Highway
Covert, MI 49043

Dear Mr. Slade:

This refers to the telephone conversation between Mr. P. M. Donnelly of your staff and Mr. M. A. Ring of this office on September 13, 1991, regarding arrangements for an enforcement conference between members of our respective organizations. This meeting is scheduled for Wednesday, October 2, 1991, 9:00 a.m. (CST), at the NRC Region III office at 799 Roosevelt Road, Building 4, Glen Ellyn, Illinois.

The purpose of this meeting is to discuss the findings of the two inspections conducted at your facility from September 19, 1990 through June 21, 1991. The inspections identified apparent violations of NRC requirements. The inspection reports, No. 50-255/90025 and No. 50-255/91202, were provided to you in NRC letters dated May 24, 1991 and August 2, 1991, respectively. Please be prepared to discuss at the Enforcement Conference in both your oral presentation and in a concise written handout the topics listed in the enclosed agenda.

If you have any questions regarding this meeting, please contact B. L. Jorgensen of my staff at (708) 790-5500.

Sincerely,

Edward G. Greenman, Director
Division of Reactor Projects

Enclosure: As stated

See Attached Distribution

RIII	RIII	Yes RIII BC	RIII RMJ for	RIII BC
Gavula/jk	Danielson	for Jorgensen	Ring	Clayton
09/20/91	09/20/91	09/20/91	09/20/91	09/20/91
yes RIII	RIII	RIII	RIII	RIII Yes
Pederson	Martin	Forney	Miller	Greenman
09/23/91	09/20/91	09/20/91	09/20/91	09/20/91

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Q PDR

SEP 23 1991

Distribution

cc w/enclosure:
David P. Hoffman, Vice President
Nuclear Operations
P. M. Donnelly, Safety and
Licensing Director
DCD/DCB (RIDS)
OC/LFDCB
Resident Inspector, RIII
James R. Padgett, Michigan Public
Service Commission
Michigan Department of
Public Health
Palisades, LPM, NRR
SRI, Big Rock Point

AGENDA

1. When considered collectively, the following apparent violations of NRC requirements represent a breakdown in the control of activities associated with piping and pipe supports:

   (a) 10 CFR 50, Appendix B, Criterion III, Design Control

   - NRC Inspection Report No. 50-255/90025; Items -01A through -01P
   - NRC Inspection Report No. 50-255/91202; Items D-2 through D-9

   (b) 10 CFR 50, Appendix B, Criterion XVI, Corrective Action

   - NRC Inspection Report No. 50-255/90025; Items -02A and -02B
   - NRC Inspection Report No. 50-255/91202; Item D-10

   (c) 10 CFR 50, Appendix B, Criterion V, Procedures

   - NRC Inspection Report No. 50-255/90025; Items -03A and -03B

   (d) 10 CFR 50.59, Changes, Tests and Experiments

   - NRC Inspection Report No. 50-255/91202; Item D-1

   For the above items, categorize and discuss the root causes and provide proposed corrective actions to prevent recurrence.

2. Compare the current design control corrective actions to the previous corrective actions taken in response to findings in IR 50-255/89007 and IR 50-255/89024 and provide an assessment of:

   (a) why previous corrective actions were not completely effective; and
   (b) why current corrective actions should be more effective than the previous actions.

3. Discuss the results of the recent independent assessment of pipe and pipe support engineering referenced in the July 9, 1991 letter from D. Hoffman (CPCo) to the NRC. Provide the action plan for implementation of any assessment recommendation.

4. Provide documentation and discuss the results of the comprehensive root cause evaluation, as committed to in the November 21, 1989 letter from K. Berry (CPCo) to the NRC and further documented in the previous enforcement board IR No. 50-255/90002. Specifically, address whether programs other than the IE Bulletin 79-14 or engineering areas other than piping and pipe support design were similarly affected by design control deficiencies.