U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
REGION IV

Report No. 99900522/80-01 Program No. 51200

Company: Bechtel Power Corporation
San Francisco Power Division
425 Market Street
San Francisco, California 94119

Inspection Conducted: March 10-13, 1980

Inspectors: C J Hale/for 3-27-80
R. H. Brickley, Principal Inspector Date
Program Evaluation Section
Vendor Inspection Branch

C J Hale/for 3-27-80
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Other Accompanying Personnel: C J Hale/for 3-27-80
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Approved by: C J Hale 3-27-80
C. J. Hale, Chief Date
Program Evaluation Section
Vendor Inspection Branch

Summary

Inspection on March 10-13, 1980 (99900522/80-01)

Areas Inspected: Design process management and follow-up on the following items reported to the NRC: unauthorized attachment of pipe support to R.P.V.; pipe support design; cable damage; error in analysis of compartment pressurization; nonconformance of bolt, stud, and raw stock material; overloading

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of structural elements due to pipe support loads; essential cooling water supply for containment air coolers; and design deficiency on main steamline snubbers and hangers. The inspection involved fifty-six (56) inspector-hours on site by two (2) NRC inspectors.

<u>Results</u>: In the areas inspected there were no deviations or unresolved items identified.

DETAILS SECTION I

(Prepared by R. H. Brickley)

A. Persons Contacted

*D. B. Hardie, Supervisor, Qu lity Engineering
*W. T. Kellermann, QA Supervisor - Programs
*F. Plutchak, Project Engineer
E. Y. Wong, Project QA Engineer

*Denotes those in attendance at the exit interview.

B. Unauthorized Attachment of Pipe Support to Reactor Pressure Vessel

This item is a follow-up to a 10 CFR 21 and 50.55(e) report by the Pennsylvania Power & Light Company (PP&L) to Region I regarding Bechtel's unauthorized attachment, by welding, of a pipe hanger assembly to the Susquehanna Steam Electric Station (SSES) Unit #1 reactor pressure vessel (PRV).

1. Objectives

The objectives of this area of the inspection were to:

a. Examine the results of the evaluation of this item to determine that a proper evaluation was performed.

b. Determine whether this item is generic or plant unique.

c. Determine if this item was properly reported to the NRC.

2. Method of Accomplishment

The preceding objectives were accomplished by an examination of:

a. Management Corrective Action Report (MCAR) No. 1-46.

b. Bechtel's "Final Report on Small Pipe Hanger Mislocation on the Reactor Pressure Vessel" dated February 1980.

3. Findings

The examination of the documents listed in B.2 above revealed:

a. The unauthorized attachment of pipe hanger SP DCA-137-H19 to the RPV was found to be due to misinterpretation of

Fabrication Isometric SP DCA-137-4H by both the Installer and responsible Field Engineer.

b. Bechtel's position is that this item would not have gone undetected for the following reasons:

(1) The pipe support would have prevented installation of the required insulation in this area.

(2) The Project Engineering stress walkdown would have found and identified this as a mislocated pipe support.

(3) The Quality Control verification of location and installation of pipe supports would identify and document it as mislocated.

(4) The Field Engineering walkdown during construction turnover would have identified and documented it as mislocated.

c. Bechtel's engineering evaluation resulted in the conclusion that this item was not reportable because:

(1) With the pipe supports as presently installed the pipe stress remains within acceptable limits.

(2) The attachment of the pipe support to the RPV with the fillet weld described, results in a heat affected zone extending approximately 1 to 2 mm into the vessel wall and the effect of the welding is a slight hardening of the material in the heat affected zone. The vessel thickness in this region is determined by matching shell courses rather than providing additional thickness as replacement for nozzle penetration. This additional thickness for this shell course results in lower hoop stresses.

Bechtel's engineering evaluation of this condition was that if the pipe support were left in place there would be no impact on the safe operation of the RPV.

d. Bechtel's corrective action is to remove the pipe support from the RPV. Their UT thickness measurements indicate sufficient excess wall to permit removal of the weld and heat-affected zone without the need for a weld repair. They anticipate completion of this repair by June 4, 1980.

e. There were no deviatons or unresolved items identified in this area of the inspection.

C. Pipe Support Design

This item is a follow-up to a 10 CFR 50.55(e) report by PP&L to Region I regarding an identified deficiency involving discrepancies between the Bechtel Plant Design Group s finalized stress analysis calculations and the loads which were used in the design of pipe support detail drawings which were fabricated and released for installation during the construction of SSES. The released pipe support detail drawings did not have factored into them revised piping system stress values which had been developed subsequent to the initial piping system stress calculations.

1. Objectives

The objectives of this area of the inspection were to:

a. Examine the results of the evaluation of this item to determine that a proper evaluation was performed.

b. Determine whether this item is generic or plant unique.

c. Determine that this item was properly reported to the NRC.

2. Method of Accomplishment

The preceding objectives were accomplished by an examination of:

a. MCAR No. 1-36 dated July 24, 1979.

b. MCAR No. 1-36, Interim Report on Pipe Support Details Load Discrepancy, SSES.

c. Bechtel IOM (Evaulation of Limerick Project Pipe Support Flow of Hanger Guidance) dated March 7, 1980.

d. Bechtel IOM (Susquehanna MCAR 36 Pipe Support Detail Load Discrepancy) dated March 13, 1980.

3. Findings

a. Bechtel's evaluation of this item resulted in the conclusion that there was a lack of adequate design change control i.e., there were no procedures in place to ensure adequate design change control between the stress analysis group and the pipe support design groups within the plant design discipline. They estimate that the number of items requiring design change is approximately 1000 pipe supports, some of which might have been overstressed beyond allowable design limits by an order of 100 percent or greater.

b. Bechtel's corrective action was to institute a Stress Isometric/ Pipe Support Verification Program covering all the Stress Isometrics and the pipe supports affected therein. In addition Bechtel's Engineering Procedures Manual, Appendix B (Plant Design Discipline) had been revised to prevent recurrence of the situation and correct the inadequacy in design change control.

c. Bechtel's investigation of this item further concluded that the situation was unique to SSES as far as active projects were concerned. In the case of closed projects (operating plants) the IE Bulletin 79-14 inspections will identify and correct these items if they exist.

d. There were no deviations or unresolved items identified in this area of the inspection.

D. Cable Damage

This item is a follow-up to a 10 CFR 50.55(e) report by PP&L to Region I regarding damage sustained by safety-related cables during cable pulling activities.

1. Objectives

The objectives of this area of the inspection were to:

a. Examine the results of the evaluation of this item to determine that a proper evaluation was performed.

b. Determine whether this item is generic or plant unique.

c. Determine that this item was properly reported to the NRC.

2. Method of Accomplishment

The preceding objectives were accomplished by an examination of:

a. MCAR No. 1-35 dated June 13, 1979.

b. Bechtel letter (Release of Cable Pulling) to PP&L dated June 25, 1979.

c. Bechtel IOM (Reportability of Cable damage) dated July 3, 1979.

d. Bechtel IOM (Response to MCARs) dated July 23, 1979.

e. PP&L letter (Cable Pulling Activities) to Bechtel dated August 9, 1979.

f. MCAR No. 1-35 Interim Report dated August 24, 1979.

g. Bechtel QA IOM (Interim Response to MCAR No. 1-35) dated September 10, 1979.

h. Bechtel IOM (Interim Response to MCAR No. 1-35) dated September 25, 1979.

i. PP&L letter (Interim Report of a Deficiency in Cable Pulling Activities (Status Update)), to Region I dated December 10, 1979.

j. MCAR No. 1-35 Final Report on Cable Damage Sustained During Cable Pulling Operations dated February 27, 1980.

3. Findings

a. The cable damage was found during cable pulling operations and QC inspection activities and was documented on NCRs. The nature of the damage was varied, and the cause could be attributed to one or a combination of eight (8) factors.

b. Bechtel's review of the NCRs and related documents indicated:

(1) Based on the June 1979 circuit schedule, more than 2,000 safety related cables are shown as pulled; damage was detected on 26 cables.

(2) The nature of the cable damage was varied, and the cause of the damage could not be attributed to one single factor.

(3) One instance of Okonite cable (1/c #4/0 neutral wire) damage was reported in NCR-3749. All other reported cable damage occurred on 600V power and control cable, #2 AWG and smaller, manufactured by the American Insulated Wire Corporation.

(4) No cable damage or overtension was discovered on any Kerite 15KV or 5KV cable, Samuel Moore instrument cable or Raychem Specialty cable.

(5) Additional testing performed on cables, per the engineering disposition of NCR-3949, verified that no cable damage existed. These had previously been reported as damaged cables.

(6) Eighteen damaged cables reported were discovered through megger tests.

Megger tests are performed on all power cables to assist in determining the integrity of installed power cables and to facilitate the pre-operational, functional testing and start-up procedures.

c. Bechtel's evaluation concluded that should any cable damage remain undetected and uncorrected during cable pulling operations, it is improbable that there would be any adverse effect to the safety of plant operations because:

(1) All installed cables are subjected to planned and scheduled tests.

(2) There is redundancy for each safety-related circuit.

(3) Cable damage was discovered and documented on NCRs.

(4) The disposition of these NCRs has been verified.

d. Based on their evaluation and jobsite cable verification program Bechtel concluded that this item was not reportable under 10 CFR 50.55(e).

e. There were no deviatons or unresolved items identified in this area of the inspection

E. Error in Analysis of Compartment Pressurization

This item is a follow-up to a Preliminary Notification issued by Region I regarding an error in the assumptions used in the Boston Edison Co. (BECo), Pilgrim Station 600 Unit No. 1, FSAR Amendment 34 analysis of compartment pressurization due to pipe breaks outside containment.

1. Objectives

The objectives of this area of the inspection were to:

a. Examine the results of the evaluation of this item to determine that a proper evaulation was performed.

b. Determine whether this item is generic or plant unique.

c. Determine that this item was properly reported to the NRC.

2. Method of Accomplishment

The preceding objectives were accomplished by an examination of;

a. Bechtel letter No. 10394-BLE-884 (Response to IE Bulletin 79-01) to BECo. dated July 3, 1979.

b. BECo. letter No. 10394-ELB-2853 (BPOC-Torus Compartment Pressurization) to Bechtel dated July 13, 1979.

c. MCAR No. 1-1 dated February 19, 1980.

d. Computer printout Program NE699, Version D2 (COPDA) dated February 7, 1980.

e. Curve of Pressure (PSIA) vs. Time (sec.) for the HPCI Line Break in Torus Compartment.

3. Findings

a. This item was identified by Bechtel during their work for BECo. in preparation of a response to IE Bulletin 79-01 wherein they identified a concern that a postulated HPCI steam line break could produce a potentially damaging collapsing pressure on the torus. The original analysis for FSAR Amendment 34 errously assumed that both reactor building truck lock roll-up doors were open.

b. Bechtel records indicate an allowable external torus pressure of 2.8 PSID and that these doors would fail at less than 1 PSID.

c. Results of a recent computer run (E.2.d and E.2.e above) indicate an external pressure of above 2.5 PSID.

d. MCAR No. 1-1 is still undergoing evaluation and has not been closed out.

e. There were no deviations or unresolved items identified in this area of the inspection.

F. Nonconformance of Bolt, Stud, and Raw Stock Material

This is a follow-up to a potential 10 CFR 21 item reported to Region V by Coast Industrial Supply Co. regarding nuts, bolts, and raw stock materials supplied to Bechtel and others wherein the material certifications were found to be invalid.

1. Objectives

   The objectives of this area of the inspection were to:

   a. Examine the results of the evaluation of this item to determine that a proper evaluation was performed.

   b. Determine whether this item is generic or plant unique.

   c. Determine that this item was properly reported to the NRC.

2. Method of Accomplishment

   The preceding objectives were accomplished by an examination of:

   a. Bechtel teletype (Suppliers Quality Alert, Coast Industrial Supply, Los Angeles, California) dated June 22, 1979.

   b. Supplier Quality Information Bulletin No. 79-2 (Bolting Material Supplied by Coast Industrial Supply Company - Los Angeles, California) dated June 28, 1979.

   c. Bechtel IOM (NCA-3800 Audit, Coast Industrial Supply Company, Vernon, California) dated June 28, 1979.

   d. Coast Industrial Supply Co. (CISCO) letter (Response to Audit) to Bechtel dated June 29, 1979.

   e. Bechtel IOM (CISCO ASME Section II Bolts and Nuts) dated August 9, 1979.

   f. Bechtel Supplier Quality Action Request (SQAR) No. 79-7 Revision 1, dated August 10, 1979.

   g. Bechtel IOM (SQAR 7-7 Status Report) dated October 2, 1979.

3. Findings

   a. This item was identified by Bechtel during their audit of CISCO on June 21, 1979, which identified five (5) program deficiencies which indicated a potential lack of traceability of ASME Section III bolting material larger than one inch in diameter furnished by CISCO. It appeared that CISCO had also furnished improperly documented bolting material to nuclear component suppliers of Bechtel.

b. In a meeting with Bechtel on June 27, 1979, CISCO representatives committed to investigate the scope of this deficiency and issue a 10 CFR 21 report as required by the NRC.

c. The only domestic project that had received this material was Midland via Pathway Bellows, Inc. and direct procurements occurring between February and December 1978. The documentation examined indicated that the Midland project had located all affected material and remedial actions were in progress.

d. There were no deviations or unresolved items identified in this area of the inspection.

e. Follow-up Item

Results of the remedial action taken on the Midland project will be examined during a future Bechtel-Ann Arbor inspection.

G. Overloading of Structural Elements due to Pipe Support Loads

This item is a follow-up to IE Information Notice No. 79-28 which was based on a report by Portland General Electric Co. (PGE) that some walks at the Trojan Nuclear Plant were found which did not have adequate structural strength to sustain the required support reactions. These inadequacies were attributed to an apparent lack of a final check of certain pipe support locations and reactions and inadequate design criteria for the reactions from supports anchored into the face of concrete block walks.

1. Objectives

The objectives of this area of the inspection were to:

a. Examine the results of the evaluation of this item to determine that a proper evaluation was performed.

b. Determine whether this item is generic or plant unique.

c. Determine that this item was properly reported to the NRC.

2. Method of Accomplishment

The preceding objectives were accomplished by an examination of:

a. Bechtel letter to the NRC dated November 15, 1979, transmitting the results of their investigation of the design control practices used in designing structures to withstand pipe hanger loads.

b. Bechtel IOM (Suggested Design Criteria for Concrete Block Walks on Nuclear Plants) dated November 30, 1979.

c. NRC letter to Bechtel dated January 7, 1980, confirming an NRC requested meeting to discuss masonary wall design and construction.

d. Bechtel letter to the NRC transmitting additional information to the document identified in a. above.

e. Bechtel IOM (January 11, 1980, meeting with NRC on Adequacy of Masonry Wall Construction) dated February 1, 1980.

f. Bechtel IOM (Palisades Plant-Block Wall Supports) dated February 27, 1980.

3. Findings

a. Bechtel's evaluation of this item was completed and transmitted to the NRC. Note: There appears to be differing opinions between the NRC staff and Bechtel regarding these walls.

b. This item appears to be generic i.e. some other Bechtel projects may be affected e.g. Palisades.

c. The documents examined indicate that the NRC was adequately informed of this item.

d. There were no deviations or unresolved items identified in this area of the inspection.

H. Essential Service Cooling Water Supply for Containment Air Coolers

This item is a follow-up to a 50.55(e) report made by Consumers Power Company to NRC Region III; stating that calculated essential service cooling water supply pressure to the containment recirculation air coolers will not meet the minimum design requirement of the FSAR (Section 6.2.2.2.3) for heat removal during accident conditions. This report further states that this reduction in cooling water pressure to the air coolers is attributed to a decreased pump suction head from decreased cooling pond differential level and additional loads being applied to the system than were originally intended.

1. Objectives

The objectives of this area of inspection were to:

a. Review and evaluate Bechtel's analysis that determined if there was an effect on plant safety, had the calculation error gone undetected.

b. Review and evaluate Bechtel's determination of root cause of the condition, and all of the necessary corrective action to preclude recurrence.

c. Review and evaluate the Bechtel analysis of 10 CFR 21 reportability, and corrective a tions taken to assure plant safety in accordance with Part 21 requirements.

2. Method of Accomplishment

The preceding objectives were accomplished by an examination of:

a. Bechtel inter-office memorandum dated March 7, 1980.

b. Bechtel inter-office memorandum dated March 3, 1980.

c. Bechtel letter to Consumers Power Company dated August 6, 1979 containing MCAR 30 interim report 1 dated August 2, 1979.

d. Bechtel final report with respect to MCAR 30 dated September 26, 1979.

e. Management corrective action report number 30 dated September 26, 1979.

f. Bechtel inter-office memorandum dated September 26, 1979.

g. Bechtel inter-office memorandum dated September 4, 1979.

3. Findings

a. General

(1) The examination of MCAR 30 final report revealed two principal circumstances contributing to the deficiency

(a) Preliminary calculations identified several errors and nonconservative assumptions, each of which contributed to the deficiency.

(b) Engineering Department Program (EDP) 4.37 design calculations had not fully been complied with. The calculation was not performed and checked in accordance with the procedure. Contrary to the EDP requirements, the calculation was not reviewed and approved. The calculation was left in a preliminary status and the results of these calculations were used for implementation of the system design and procurement of the service water pumps.

(2) Upon examining MCAR 30 final report potential safety implications and evaluation, it states: "Following a MSLB or LOCA, the containment pressure and temperatures decrease more slowly when one spray train is used rather than one spray and one air cooler train. The temperature and pressure environment qualification envelopes of various components inside the containment may be exceeded for LOCAs and large MSLBs. Therefore, the longer term availability of various safety-related equipment and post-accident monitoring components cannot be ensured." Bechtel concluded that this item is reportable in accordance with 10 CFR 50.55(e). Bechtel's disposition of Part 21 reportability is still in the evaluation state.

(3) Upon examining MCAR 30 final report corrective action; this section states: "A pump is being provided in each service water supply line to the CRACVs to boost the service water pressure to a minimum of 40 PSIG at the outlet of the coolers . . . . Design for this corrective action is essentially complete. IDCN 69, issued August 30, 1979, shows the change to be implemented on P&ID 7220-M-419(Q)." It also states: "MED 4.37 was revised (Revision 10) to clarify that calculations are to be checked and approved prior to the use of their results in finalizing a design basis."

b. Deviations and Unresolved Items

None identified.

c. Follow-up Items

An examination of the results of the final evaluation, still in progress, will be conducted during a future inspection.

I. Design Deficiency on Main Steamline Snubbers and Hangers

This is a follow-up to an item reported to Region IV by Arkansas Power & Light that transient/dynamic loads from closure of the main steam stop valve were not considered in the design of these snubbers and hangers on the main steam line between the steam generators and the reactor building wall of Arkansas unit number one. This review of dynamic design loads included eight snubbers and two ridge hangers.

1. Objectives

   The objectives of this area of inspection were to:

   a. Review and evaluate Bechtel's analysis that determined the safety implications had this deficiency in design gone unchecked.

   b. Review and evaluate Bechtel's design review interface with the licensee and the subsequent design change implementation.

   c. Review and evaluate Bechtel's current corrective action on the deficiency.

2. Method of Accomplishment

   The preceding objectives were accomplished by the examination of:

   a. Management corrective action report number two dated July 18, 1979.

   b. Bechtel inter-office memo dated August 22, 1979.

   c. Bechtel inter-office memo dated September 4, 1979.

   d. Bechtel letter to Arkansas Power & Light dated July 16, 1979 (BL-G-934).

   e. Bechtel Engineering Department procedures EDP 4.74 Revision 0, dated March 7, 1975.

3. Findings

   a. General

      (1) The examination of MCAR 2 final report dated July 18, 1979, states: Under the heading of Description of Deficiency:

         (a) "In May of 1974 the Piping Stress Group discovered that the dynamic load must be included in the analysis, and an IOM dated May 15, 1974, was transmitted to the Arkansas Project requesting review of the design based on the dynamic load. For unknown reasons, although the memo was received, no action was taken to review the design."

(b) Under the heading safety implications it states: "The resultant effects of a steam line failure have been analyzed and reported in the ANO-1 FSAR, Section 14.2.2 this analysis has been reviewed and approved by the NRC (AEC) as it indicates that the health and safety of the public would not be jeopardized by such a failure."

(c) Recommended Action states: MCAR 2 dated July 18, 1979.

    (1) Review all seismic Category I main steam pipe support restraints for adequacy considering steamhammer transient forces (done as of July 6, 1979).

    (2) Perform analysis of safety implications to plant operations had the condition gone uncorrected (done as of July 16, 1979).

    (3) Provide design modifications to correct the noted deficiencies in the pipe supports/restraints and the over stress areas of the pipe. (In process.)

    (4) Walk down the piping and support/restraint areas which were determined to be deficient and identify any sign of abnormality. (Done as of July 16, 1979.)

(d) The examination of Engineering Department Procedure 4.74 satisfied the objective of determining Bechtel's design interface with the licensee. This interface is in the form of problem alerts which are sent from Office Chief Engineers to the involved projects. It states; "Copies of problem alerts shall be used by the cognizant Thermal Power Management Chief Engineer in the preparation of modification of the appropriate thermal power organization design guides, standard specifications or other Generic Design Documents."

b. Deviations and Unresolved Items

None identified.

J. Exit Interview

An exit interview was held with management representatives on March 13, 1980. In addition to those individuals indicated by an asterisk in paragraph A of each Details Section, those in attendance were:

P. Becnel, Counsel
H. Friend, Manager, Division Engineering
D. W. Halligan, Vice President and Deputy Division Manager
S. I. Heisler, Manager, Division QA
S. M. Jenks, Division Management Staff
T. M. Leverette, Project Manager

The inspector summarized the scope and findings of the inspection. In addition, the inspector reminded, the management representatives present that responses to deviations identified during future inspection reports should contain, as a minimum the following:

1. Corrective Actions

A description of the steps that have been or will be taken to correct the item, the steps that have been or will be taken to assure that similar items do not exist, and the date these actions were or will be completed.

2. Preventive Measures

A description of the steps that have been or will be taken to prevent recurrence of this type deviation and the date these preventive measures were or will be completed.

In both cases the corrective and preventive actions must be documented and capable of being verified by the NRC inspector during a subsequent inspection.

Management comments were generally for clarification only or acknowledgement of the statements by the inspector.

DETAILS SECTION II

(Prepared by J. R. Agee)

A. Persons Contacted

E. J. Gough, Electrical Group Supervisor
M. J. Jacobson, Project Quality Assurance Engineer
L. A. Johnson, Control Group Supervisor
J. M. Lenschau, Project Administrator
J. A. McCall, Mechanical Group Supervisor
W. P. Neuendorf, Nuclear Engineer
S. Proroczok, Project Quality Engineer
*B. N. Pusheck, Project Engineer

*Denotes those present at the exit interview.

B. Design Process Management

1. Objective

The objective of this area of inspection was to examine the establishment and implementation of quality related procedures for the design process to verify that:

a. The design process system is defined, implemented, and enforced in accordance with approved procedures, instructions, or other documentation for all groups performing safety related design activities.

b. Design inputs are properly prescribed and used for translation into specifications, drawings, instructions, or procedures.

c. Appropriate quality standards for items important to safety are identified, documented, and their selection reviewed and approved.

d. Final design can be related to the design input with this traceability documented, including the steps performed from design input to final design.

e. Design activities are documented in sufficient detail to permit design verification and auditing.

f. The methods are prescribed for preparing design analysis, drawings, specifications, and other design documents so that they are planned, controlled, and correctly performed.

2. <u>Method of Accomplishment</u>

The preceding objectives were accomplished by an examination of:

a. Section 3 (17.1.3) Design Control, Topical Report BQ-TOP-1, Revision 2A, dated July 1977.

b. The following sections from the Quality Assurance Policy Manual:

   (1) Section II, No. 2, Design Control Procedures, Revision 2, dated April 14, 1978.

   (2) Section II, No. 4, Design Criteria, Revision 1, dated March 1, 1974.

   (3) Section II, No. 5, Design Process and Change Control, Revision 1, dated March 1, 1974.

c. Engineering Department Project Instruction (EDPI), Design Interface Control (Job 8791) - 4.24.1, Revision 4, dated December 12, 1978.

d. Emergency Feedwater System Description, 8791-SDM-23, Revision 0, dated October 29, 1976.

e. Technical Specification for Containment Post-LOCA Hydrogen Monitoring System, Specification No. 8791-J-359, Revision A, dated April 28, 1978 and discussions with the nuclear engineer responsible for nuclear shielding which revealed this system may be revised to incorporate radiation criteria to provide radiation protection levels to that which could have been seen in the containment vessel following the Three Mile Island incident. A determination of the need for increased radiation criteria will be made upon completion of current applicable radiation studies that are in progress.

f. Technical Specification for 4.16KV Metal-Clad Switchgear Class 1E (Safety Related), 8791-E-809, Revision 0, dated October 17, 1977, which revealed that this specification and its referenced equipment qualification specifications required compliance to the latest revisions of codes and standards that are addressed in the Safety Analysis Report (SAR) Section 3.10, Design of Category I Instrumentation and Electrical (Class 1E) Equipment, and Section 3.11, Environmental Design of Mechanical and Electrical Equipment.

g. The procurement material requisition MR-E-888, Revision 0, for 125V, DC Distribution Bus and Panels, Class 1E (Safety Related) which had been issued for bid.

h. The following equipment drawings: E-2017(Q) and E-2005, which revealed that each had been drawn, reviewed, checked, approved and distributed according to the Nuclear Quality Assurance Manual (NQAM), Design Control Procedures, Transmittal and Distribution of Design Documents. The inspector also examined EOP 5.25 Project Master Distribution Schedule, Revision 1, dated October 18, 1974, which provides the instructions for establishing the Project Administrative Department Procedures (ADP) for identifying, reproduction and distribution of project documents.

3. Findings

a. By examination of the Emergency Feedwater System and the related equipment specifications for the 4.16 KV switchgear and the Containment Post-LOCA Hydrogen Monitors, the inspector verified that design criteria, referenced codes and standards, design interfaces, change control, format and preparation of engineering procedures, distribution of project documents described and/or discussed in the documents referenced in paragraphs a. through c. above, were adhered to in the preparation of this system specification.

b. The project nuclear engineer stated that as a result of the Three Mile Island incident additional radiation studies are in progress regarding radiation criteria for certain project equipment, e.g. 4.11 KV switchgear located in zone 2 of the containment building. Radiation criteria for this equipment may be increased by as much as fifty percent (50%) which will require revision of applicable sections of the SAR and related switchgear specifications. Similar changes may be made to other systems and equipment specifications as a result of these radiation studies.

c. Paragraph 4.0 of Unresolved Matters of the Emergency Feedwater System Description 8791-SDM-23, identified several items that require further resolution before the system description can be considered final. Other specifications that were examined contained similar evidence of "unresolved matters." A separate review of Bechtel, SFPD engineering design standards, revealed that the use of "Unresolved Matters" in a specification is used in the initial and intermediate issues wherein items are identified for ultimate inclusion, but for which specific details are not known at the time of the issue. This section (Unresolved Matters) will grow progressively shorter and contain only the word "concluded" as the specific required details are resolved.

d. Within this area of the inspection no deviations or unresolved items were identified.