SMUD
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AN ELECTRIC SYSTEM SERVING THE HEART OF CALIFORNIA

RJR 85-385

August 6, 1985

AUG 6 REC'D

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U S NUCLEAR REGULATORY COMMISSION
REGION V
OFFICE OF INSPECTION AND ENFORCEMENT
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DOCKET 50-312
RANCHO SECO NUCLEAR GENERATING STATION
UNIT NO. 1
MANAGEMENT REPORT CONCERNING JUNE 23, 1985
HIGH POINT VENT LEAK AT RANCHO SECO

Attached for your information is a report from District Management to the Management Safety Review Committee (MSRC) addressing the concerns arising from the June 23, 1985 high point vent leak at Rancho Seco. This report was presented to the MSRC on August 5, 1985.

District Management has directed the completion of the following items prior to restart of the plant:

- Disposition all nonconformance reports (NCR) generated during the walkdown activities discussed in the management report.
- Complete all rework required by the NCR dispositions.

To date, the District has completed the following:

- Evaluated the B loop high point vent pipe crack and piping system.
- Evaluated whether similar concerns exist for other systems including:
  - Post 79-14 walkdown
  - Supplemental 79-14 walkdown
  - NRC audit 85-01 walkdown results with respect to conduit, cable trays, and HVAC supports
- Repaired B loop high point vent piping and supports.

IE-01
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- Documented nonconformances identified by walkdowns.

- Repaired and tested the Nuclear Services Cooling Water pump breaker.

- Evaluated the effectiveness of the Rancho Seco surveillance program for the following systems:

  - Auxiliary Feedwater System
  - High Pressure Injection System
  - Decay Heat/Low Pressure Injection System
  - Reactor Protection System

- Evaluated and corrected the diesel generator control circuit.

- Implemented an Equipment Environmental Qualification Program.

- Upgraded the Steam Generator Tube Inspection Program.

By September 21, 1985, the District will provide you with a completion schedule for the longterm items discussed in the report. If you have any questions feel free to contact Robert Little of my staff at (916) 732-6021.

R. J. RODRIGUEZ
ASSISTANT GENERAL MANAGER,
NUCLEAR

Attachment

Return to L. Miller

EVALUATION OF CONCERNS

ARISING FROM THE JUNE 23, 1985

RCS HIGH POINT VENT LEAK

AT THE

RANCHO SECO

NUCLEAR GENERATING STATION

EXECUTIVE SUMMARY

The Sacramento Municipal Utility District is committed to strive for excellence in the operation of the Rancho Seco Nuclear Generating Station. In support of this commitment, the District has been in the process of critically reviewing its operating practices and making programmatic improvements. These improvements are far reaching and affect virtually all aspects of the District's nuclear operations from design through plant operation. A few typical improvements are listed below:

- Management consultants reviewed the District's nuclear organization and suggested changes to strengthen the nuclear operation. The District is acting to implement the recommendations including a critical self appraisal of those actions.

- The District has embarked upon a program to increase its technical self-sufficiency by augmenting its technical staff and decreasing its dependence on contractor services.

- The large majority of the technical staff is located at the plant site to facilitate design implementation and improve communications.

- The District is currently formalizing and revising its design and construction procedures to ensure attention to detail and the use of proper design and construction practices.

- The District is implementing a root cause investigation program to ensure that the root causes of problems are determined, that lessons are learned from problems, and to minimize the possibility for recurrence of problems.

- The District is purchasing a simulator to provide additional training and technical capability for the Rancho Seco staff.

The District's response to recent events is indicative of the management's attitude toward excellence in the operation of Rancho Seco. The District has taken action to ensure that the plant is safe and that it remains safe throughout its operating life.

On June 23, 1985, Rancho Seco operators discovered a leak in the reactor coolant system. Upon further investigation, the District identified the leakage location as a crack in the "B" loop hot leg high point vent piping. By comparing the design drawings for the piping system with the as built condition, Nuclear Engineering discovered that configuration differences between the design and as built piping existed. Further evaluation by Nuclear Engineering led them to conclude that the configuration differences contributed to the pipe crack. In response to this discovery, the Management Team decided that a walkdown of other safety related pipe support systems should be performed to ensure that the plant was in a safe configuration. All safety related piping modified since the walkdown performed in response to I&E Bulletin 79-14 was included in the walkdown scope.

The Management Team reviewed the scope of the original 79-14 walkdown and decided to expand it to include the appropriate Class I safety related piping which was not originally addressed by the 79-14 walkdown.

EXECUTIVE SUMMARY (Continued)

The walkdowns, which are now complete, identified nonconformances or deviations from the design drawings. The engineering walkdown teams prepared nonconformance reports (NCRs) to address the discrepancies. The District has evaluated all nonconformances to determine acceptability. As a result of this evaluation, the District will rework approximately 250 nonconforming supports to restore the intended design margins. Prior to the restart of the plant, the District will complete all rework using the District's standard inspection practices. In addition, all rework will be reinspected by the walkdown teams to ensure configuration conformance.

As a result of the walkdown process, and other ongoing reviews, the District has identified a number of programmatic changes to minimize the potential for recurrence of configuration problems. The District is continuing the programmatic review process and will implement further improvements as they are identified. The programmatic changes identified below will be completely implemented by October 1, 1985:

- Implement Design Guides (new nuclear engineering procedures). Stress in the training of design personnel the need to indicate tolerances on design drawings.

- Implement the new construction engineering procedures/inspection standards and provide training to appropriate personnel.

- Improve training of craft supervision with respect to installation in accordance with design drawings.

- Assure that the training for field engineers addresses verification of the installation.

- Train QC inspection personnel emphasizing that inspection and acceptance of work must be performed to approved design documents. (Initial training for QC inspectors was accomplished during initial phase of NRC Audit 85-01 Investigation.)

- The District has reorganized to combine two separate plant QC groups into a single division and placed that division into the QA Department.

- Continue to implement the QA surveillance program for modification work which was reemphasized in the first quarter of 1985 with the issuance of scheduled QA surveillances.

- Investigate methods to enhance the tracking of NCRs.

- Provide procedural controls for removing structural support components.

District management has suspended the majority of all Class I new construction modification work until the implementation of the above programmatic changes and training is completed. The exception is any essential Class I work which will be subject to a separate configuration verification by design engineering.

EXECUTIVE SUMMARY (Continued)

The District has also considered the generic implications of the high point vent pipe failure on the plant-wide support systems for conduit, cable tray and HVAC. An engineering team performed a walkdown of a random sample of plant safety related support systems in response to NRC Audit 85-01. The results of this walkdown indicated that although minor deviations from design existed in the plant support systems, none of the deviations were found to exceed design allowables. This evaluation has lead the District to conclude that the other plant support systems are safe.

The District's commitment to strive for excellence in the operation of Rancho Seco is also apparent in the District's ongoing programs and responses to other recent events such as:

- The District formed a task force to evaluate the nuclear service cooling water pump breaker failure and determine and implement corrective actions.
- The District promptly evaluated and corrected the diesel generator circuit problem.
- The District immediately investigated the reactor trip breaker failure and worked with the vendor and the B&W Owners' Group to resolve the issue.
- The District sent a representative to the Davis-Besse site following the loss of feedwater incident to gain firsthand knowledge of the event specifics. The District formed a task force to ensure that lessons learned from the Davis-Besse incident are implemented at Rancho Seco.
- The District has developed an extensive equipment environmental qualification program which recently received a favorable NRC audit.
- The District has implemented an extensive Appendix R program.
- The District developed a program to evaluate the bus bar bronze bolt failures.
- The District implemented new OTSG tube inspection criteria.
- The District has initiated betterment improvements to nuclear fuel handling equipment.
- The District assembled a task force to evaluate and implement solutions to the meteorological tower problems.
- The District has evaluated the effectiveness of the Surveillance Program and is implementing program improvements.
- The District is developing a trending program to track surveillance test results.

EXECUTIVE SUMMARY (Continued)

In conclusion, the District has taken immediate action to ensure that the Rancho Seco Nuclear Generating Station is in a safe operating condition. These actions include:

- The repair of the "B" loop high point piping and supports;
- The disposition of all NCRs related to the pipe support walkdown effort;
- The completion of all support rework prior to restart; and
- The repair and testing of the nuclear service cooling water pump breaker.

These actions, coupled with the District's long term programmatic changes and commitment to strive for excellence, ensure that the Rancho Seco Nuclear Generating Station will continue to be operated safely in the future.

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I INTRODUCTION

The Sacramento Municipal Utility District (the District) is committed to strive for excellence in the operation of the Rancho Seco Nuclear Generating Station. In support of this commitment, the District has been in the process of critically reviewing its operating practices and making programmatic improvements in the areas of design control, modification control, training, and organization. Recent events at Rancho Seco have been adressed in a manner which demonstrates the District's commitment to strive for excellence.

This report describes the District's evaluation of the June 23, 1985, high point vent pipe crack and provides an assessment of the effect of the concerns identified during that evaluation on the plant-wide systems. This report also enumerates and discusses the District's actions to ensure that the plant is in a safe configuration and remains in a safe configuration throughout its operating life.

In addition, the District's response to other events and ongoing District programs are presented. These responses and programs are indicative of the District's commitment and management attitude that Rancho Seco will not only be operated safely, but that operations will continue to improve.

II RCS PIPE CRACK DURING JUNE 23, 1985, PLANT STARTUP

On June 23, 1985, with the plant in a hot shutdown condition, the Rancho Seco Control Room Operators detected a leak from the Reactor Coolant System (RCS). The leakage rate was estimated at approximately 17 gpm and determined to be from a crack in an unisolable segment of the 1" diameter "B" RCS hot leg vent. (The "B" RCS hot leg high point vent is connected to the plant vent header, the nitrogen supply header, and the emergency high point vent to the reactor building atmosphere.) The plant was subsequently cooled down and depressurized so that the RCS could be drained down and the pipe repaired.

A. Investigation of Failure Mode

Visual inspections by Nuclear Operations determined that a through-wall crack was located on the pipe side of a pipe to tee weld shown on Figure II-1. The crack extended approximately 120° around the pipe circumference on the west side and seemed to follow the weld fusion line. An investigation was initiated to try to determine the failure mode for the cracked pipe. The investigation included a walkdown of the "B" hot leg high point vent to document the as found piping configuration, stress analysis of the as found configurations, and removal of the segment of the pipe and tee with the crack for laboratory examination.

The pipe segment with the crack was shipped to General Electric Co. (GE), Vallecitos Nuclear Center, for examination. GE found that the cracking was transgranular and, although there was no direct metallurgical evidence of fatigue crack propagation, GE concluded that the most likely cause of the cracking was high cycle fatigue. Attachment I, Metallurgical Evaluation of Rancho Seco Steam Generator Vent Line Cracking, contains the details of GE's evaluation. Consultants from both Bechtel Power Corporation and IMPELL inspected the cracked pipe and reviewed GE's finding and concurred with GE's metallurgical evaluation.

The District compared the as found configuration, shown on Figure II-1, with that of a 1983 drawing change notice (DCN), shown on Figure II-2, and found that three piping supports and a dummy spool piece which were considered in the design stress analysis were not installed. Nuclear Engineering then performed a stress analysis on the as found configuration. Table II-1 shows the as found piping stresses compared with the piping stress for the current design shown on Figure II-3. The data shows that under normal conditions, the combined normal primary stresses (dead weight plus pressure) in the area of the crack (data point 5) do not exceed code allowable limits. However, the seismic stresses for the as found configuration exceed the allowable limits. The dead weight analysis, however, revealed that there was an unusually large deflection (0.83 inches) at the blind flange end.

Since there had not been any seismic activity at Rancho Seco, the as found stress analysis results indicate a cyclic fatigue induced failure. To strengthen this conclusion, Nuclear Engineering performed a simplified fatigue evaluation.

A. (Continued)

Nuclear Engineering first calculated the allowable pipe stress levels for steady-state vibration using the ANSI/ASME OM3-1982 methods for ASME Class 2 and 3 piping.

By working backwards, the analyst was able to calculate the acceleration value required to produce stresses that would equal the allowables. The results of the evaluation show that acceleration levels greater than 0.633 g can produce stress at the nozzle connection that exceed the stress limits for steady-state vibration. The calculated (0.59 inches) deflection of the piping and was found to be less than actual (0.83 inches) deflection of the piping due to its own weight and other in line components. The predominant frequencies of the cantilevered piping were calculated to be 3.32 cps in the vertical direction and 8.49 cps in the horizontal direction. These frequencies are within the range of flow induced vibrations observed at another B&W plant.

Two other observations with regard to the crack location were noted. First, the crack occurred in the heat affected zone of the weld which inherently has significant residual stresses. Second, the crack occurred at a location which has high longitudinal tensile stresses compared to compressive stresses on the intact side of the pipe. Using these two factors, the results of the fatigue calculations, and the absence of conditions leading to another failure mechanism, the District concluded that a fatigue induced failure of the "B" hot leg high point vent line was the most probable cause.

B. Break Scenario Evaluation

As a result of the June 23, 1985, high point vent line leak, District management directed the Rancho Seco Shift Technical Advisor (STA) Group to prepare an STA Transient Assessment Report (TAR) as a reading assignment for the Rancho Seco operating staff (see Attachment II). At the time of the event the reactor was in a hot shutdown condition when the operators received indications of a leak in the primary system. After investigating the problem and determining the location and type of leak, the operators began a normal plant cooldown per Operating Procedures "Plant Shutdown and Cooldown." The cooldown proceeded normally and, during the transient, no situation arose in which use of the Emergency Operating Procedures (EOPs) was required, with the exception of the declaration of an Unusual Event. Throughout the cooldown, the makeup system was able to maintain pressurizer level and RCS pressure. Likewise, subcooling margin was also maintained and T_{avg} did not increase.

The transient ended with the operators, as well as plant systems, performing as intended.

B. (Continued)

However, due to the location of the RCS leak and the potential for a complete pipe break, District management requested further analysis. This consisted of expanding the scenario to ensure that the plant could be shutdown in a safe and orderly manner had this been a complete rupture of the 1" schedule 160 high point vent line.

As noted in the STA's TAR, several scenarios were briefly examined indicating that with normal operator response, the Safety Features Actuation System (SFAS) would <u>not</u> automatically initiate. The District then decided to explore other 1" break scenarios assuming different initial conditions, (hot shutdown and 100% full power) with and without (for 10 minutes) operator action, and with one HPI pump unavailable. The District decided on four representative scenarios (see Table II-2) that were subsequently performed on the Babcock and Wilcox (B&W) simulator in Lynchburg, Virginia.

Attachment III details the sequence of events and results of the four simulator runs. It is evident, that with operator action, automatic initiation of SFAS would not occur. Assuming no operator action for 10 minutes, SFAS initiates, but the operator does regain control of plant conditions within 6 minutes of initiating operator action. In either case, plant safety systems will mitigate the consequences of a 1" line break and allow the reactor to be shut down in a safe and orderly manner. In a realistic scenario, the operator would take action to control the shutdown and the Safety Features System would not be challenged.

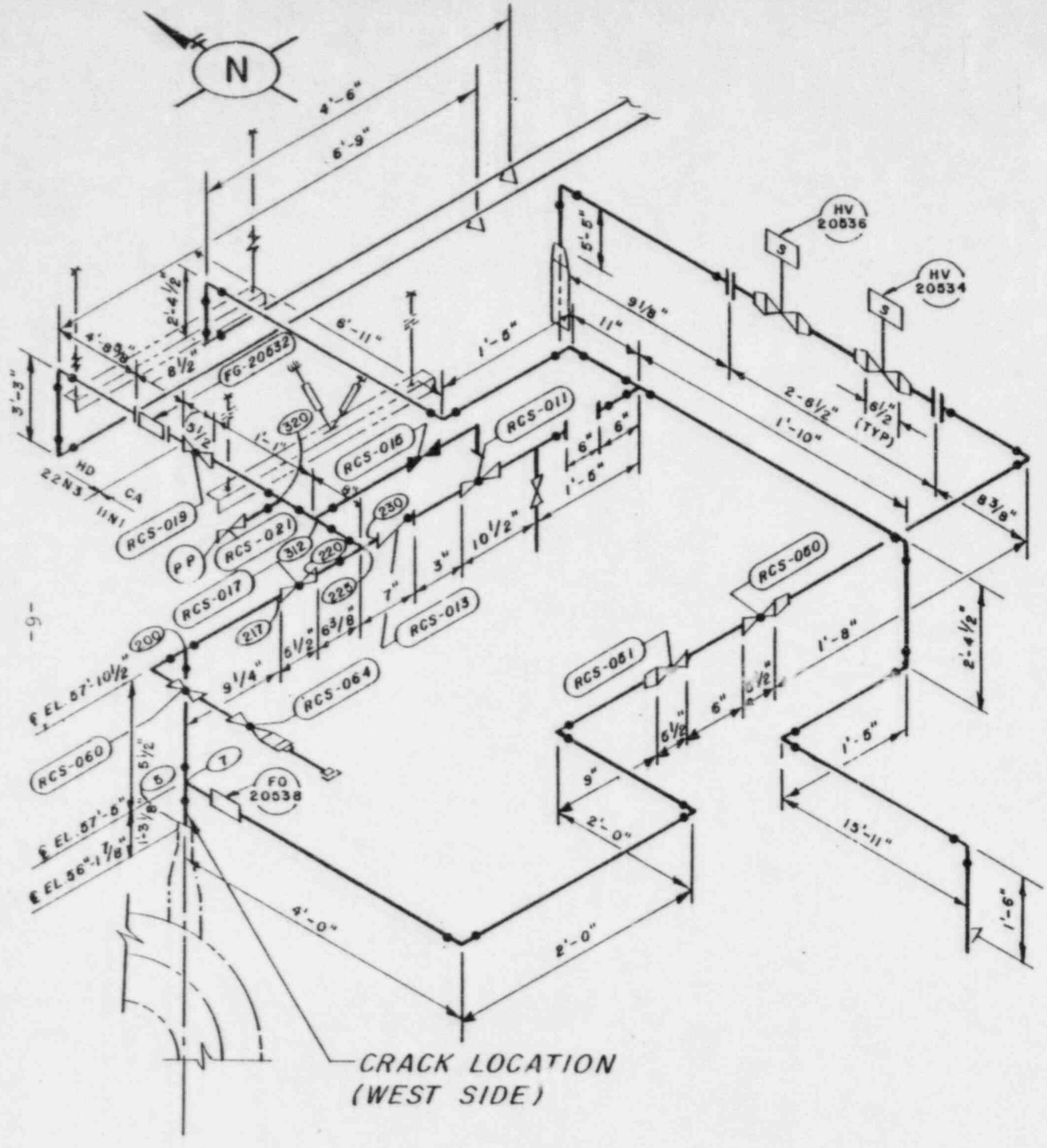
The simulator runs are consistent with the B&W small break LOCA (SBLOCA) analyses performed as required by 10 CFR Appendix K. Figure II-3 and Table II-3 show a spectrum of break sizes. Since the high point vent line (1" schedule 160 pipe) has an internal diameter of 0.815 inches, it falls within the low end of the SBLOCA region.

Furthermore, as part of their training, operators are taught that their actions can have a significant influence on the rapidity of recovery from an SBLOCA event. This is stressed in their training and during their simulator classes and is borne out in the four scenarios performed for this event. In the case of the June 23, 1985 event, the operators had indications of an RCS leak and were prepared to act accordingly had there been the complete rupture of the high point vent line.

In conclusion, the TAR, the four simulator runs, and the B&W SBLOCA analysis all confirm that had the 1" high point vent line ruptured, the plant would have been shutdown in a safe and orderly manner. This conclusion is applicable to all conditions the plant was in during Cycle 6 up to Cycle 7 startup including hot shutdown and full power, with indications of a leak or with a sudden rupture of the 1" line and with normal operator action or with no operator action for 10 minutes.

C. Repair of Piping and Supports

Since the District determined, as a result of investigation, that the probable failure mode was cyclic fatigue induced failure, there was a concern as to the remaining fatigue life of the rest of the highly stressed system piping. To eliminate this concern, the District replaced the piping from the "B" hot leg vent nozzle to orifice FO-20530 and valves RCS-060, RCS-(RCS-015, and RCS-019. The District redesigned and thereby imp ved the the piping configuration as shown in Figure II-4. The redesign minated the small piece of pipe between the hot leg nozzle and the (the segment where the crack was located) and a piece between the tee nd valve RCS-015, two welds and the supports at RCS-015. The remaining modifications required by the 1983 DCN were also made since the new configuration did not significantly change from the 1983 DCN. The north support was modified to a rigid support and the south support was modified to provide rigid lateral bracing. Also, a rigid dummy spool piece was installed in the nitrogen supply piping.



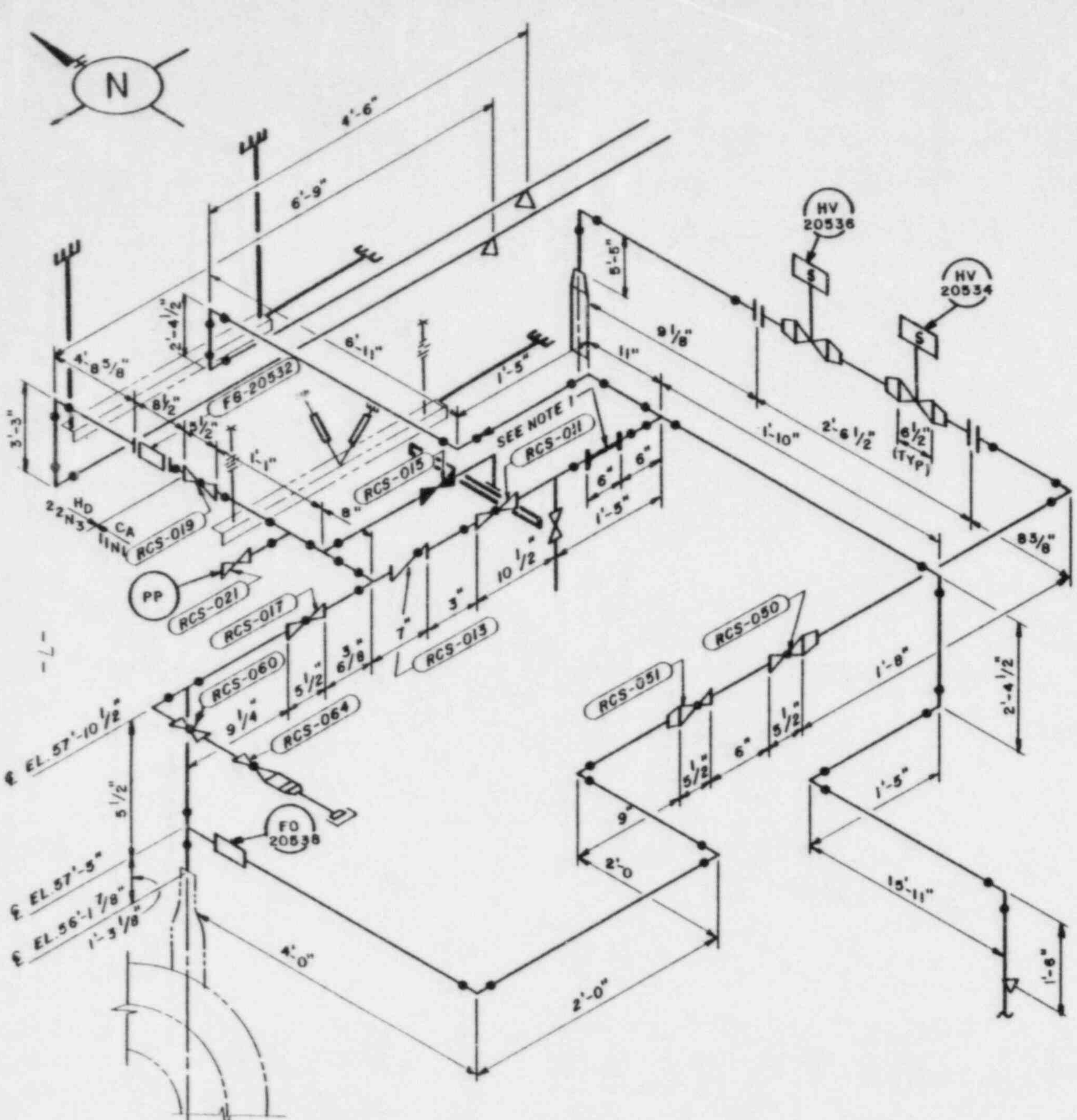
AS FOUND

ISOMETRIC DIAGRAM

R. C. S. HIGH POINT VENT
STEAM GENERATOR E-205B

20564 - 1" - HD

FIGURE II - I



AS REQUIRED BY 1983 DCN PACKAGE

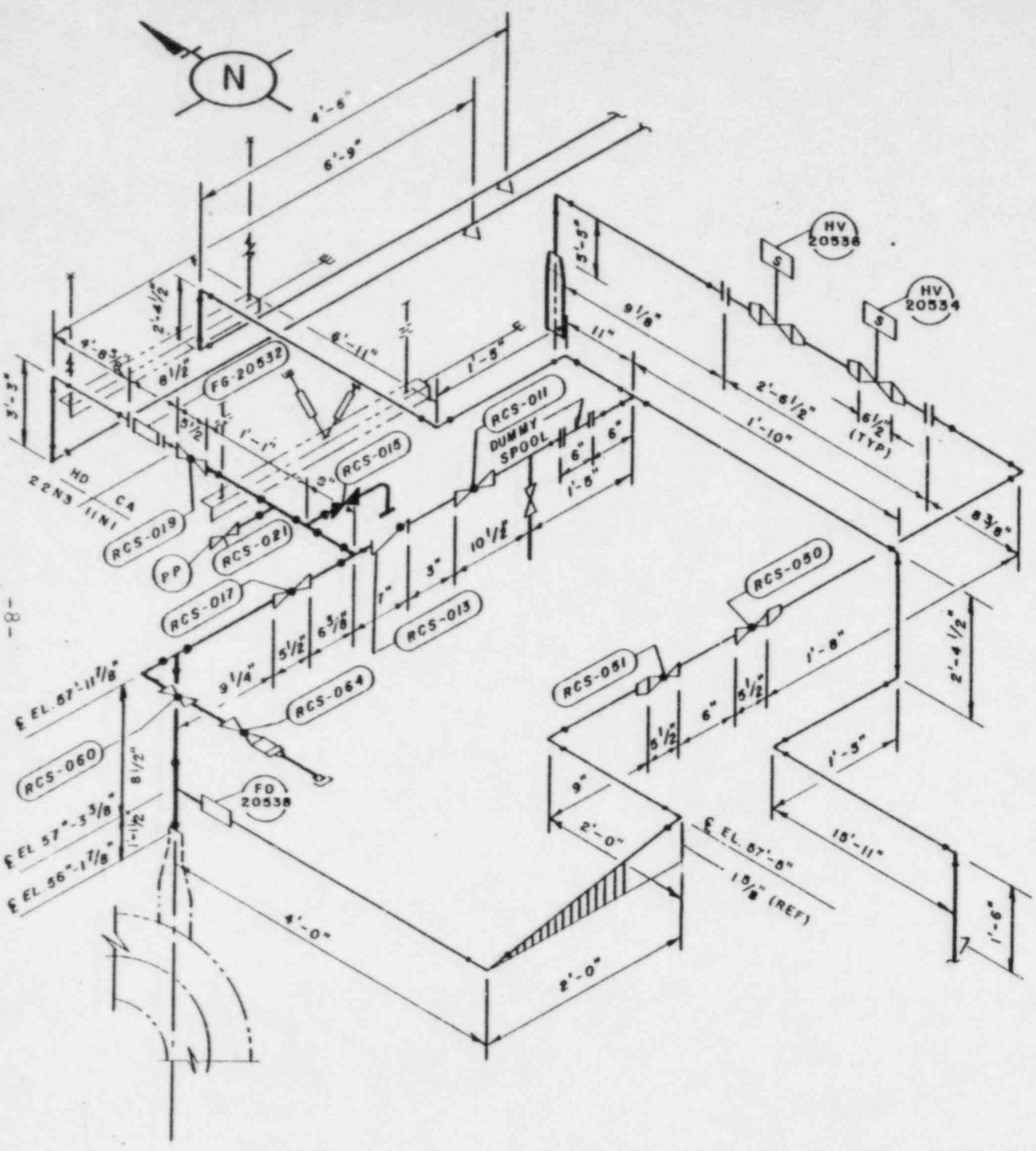
NOTE:

1. FIELD TO WELD RIGID MEMBER BETWEEN TWO FLGS.

ISOMETRIC DIAGRAM
R.C.S. HIGH POINT VENT
STEAM GENERATOR E-205B

20564 - 1" - HD

FIGURE II - 2



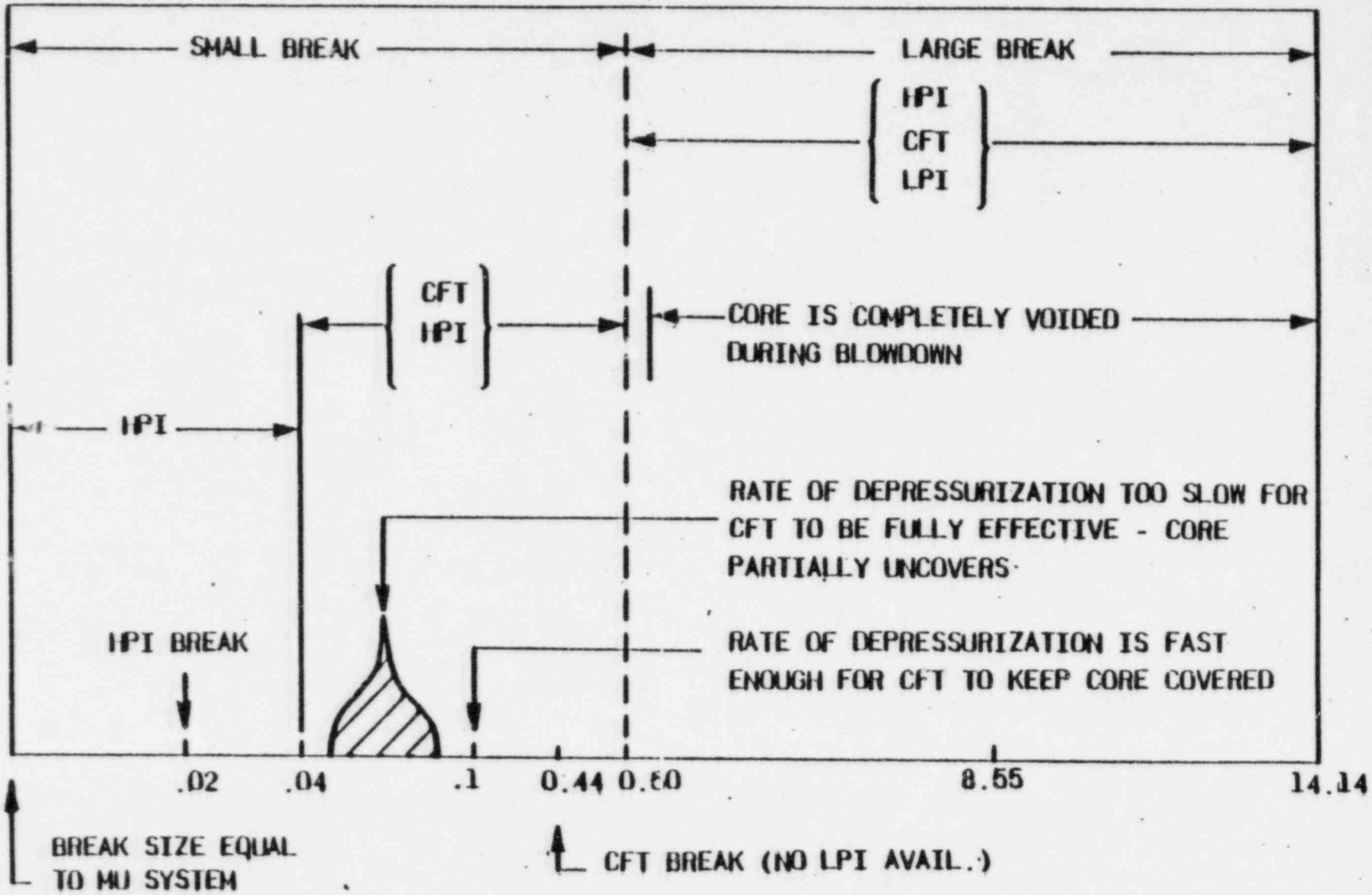
-8-

AS MODIFIED

ISOMETRIC DIAGRAM
R.C.S. HIGH POINT VENT
STEAM GENERATOR E-205B
20564 - 1" - HD

FIGURE II - 3

ECCS Vs Break Size



BREAK SIZE FT^2

FIGURE II-4

SBLOCA(177 FL)-37

R.C.S. "B" VENT

DATA PT.	COMPONENT	PRESS. STRESS PSI	WEIGHT STRESS PSI	SEISMIC STRESS PSI (OBE)	THERMAL STRESS PSI
5	NOZZLE	1559	7065 / 2708	26295 / 4320	15850 / 14431
7	TEE (RUN)	1559	6762 / 3421	9484 / 1708	16835 / 14840
200	TEE (BRANCH)	1559	6367 / 2844	10845 / 1866	12943 / 20805
217	VALVE END S.W.	1559	7346 / 1799	18005 / 3020	21337 / 36236
220	VALVE END S.W.	1559	6485 / 1005	20683 / 3602	18934 / 33192
225	TEE (RUN)	1559	7439 / 1351	19739 / 2215	8281 / 13718
230	VALVE END B.W.	1559	7232 / 1728	19802 / 2245	— / 6043
312	BRANCH CONN.	1559	4511 / 2942	11529 / 2192	4554 / 8610
320	TEE (RUN)	1559	8709 / 2312	10955 / 1957	3384 / 7522

LEGEND

- Upper value: WITHOUT DUMMY SPOOL (AS-FOUND COND.)
- Lower value: WITH DUMMY SPOOL (MODIFIED COND.)

TABLE II-1

SIMULATOR SCENARIOS

1) HOT SHUTDOWN - 535°F, 2,155 psig - preliminary indications of 17 gpm leak.
- One HPI pump unavailable

As operators start to cool down/depressurize - leak becomes 86.6 lbm/sec.
STOP when pressurizer level and RCS pressure are stable.

2) HOT SHUTDOWN - 535°F, 2,155 psig - 1" pipe break (guillotine) (86 lbm/sec)
- One HPI pump unavailable

No operator action for <u>10 minutes</u>.
STOP when pressurizer level and RCS pressure are stable.

3) 100% FULL POWER - 582°F, 2,155 psig
- One HPI pump unavailable

Preliminary indications of 17 gpm that increases to 53.3 lbm/sec (1" Schedule 160 pipe break at specified temp and pressure).
STOP when pressurizer level and RCS pressure are stable.

4) 100% FULL POWER - 582°F, 2,155 psig
- One HPI pump unavailable

1" Schedule 160 pipe break - leak rate of 53.3 lbm/sec, then escalate leak rate to 86.6 lbm/sec on Reactor Trip.

No operator action for <u>10 minutes</u>.
STOP when pressurizer level and RCS pressure are stable.

TABLE II-3

BREAK SIZE CONVERSION

Area (ft^2)	Diameter (in)	
.00065	.35	O Leak
.0014	.5	o SBLOCA
.007	1.13	(PORV)
.01	1.35	
.025	2.14	(Pzr CSV)**
.07	3.58	
.1	4.28	
.2	6.06	
.5	9.57	O SBLOCA
*8.55	28	o LBLOCA

* 2-area double ended break (cold leg)

** Pressurizer code safety valve

NOTE: Because of possible seam breaks, splits, partial pipe separations, any break size is possible.

III APPLICABILITY TO PLANT-WIDE SYSTEMS

During the investigation of the "B" loop high point vent pipe crack the District discovered that portions of the high point vent system were not in conformance with the design documents. In response to this discovery, the District's management directed that further investigations be performed, in addition to ongoing District programs, to determine applicability of the configuration concerns to the plant-wide systems.

A. Post 79-14 Walkdown of Pipe Supports

Review of the"B" RCS high point vent line crack led to the decision by the District to systematically inspect safety related Class I pipe supports. The scope of the inspection covered all safety related Class I supports installed since the as built verification required by IEB 79-14. A detailed inspection procedure was written (Attachment IV) to ensure a consistent approach to the inspection and to document the walkdown criteria. Special attention was given to piping with removable spool pieces similiar to the "B" RCS hot leg vent nitrogen supply piping.

In accordance with the inspection procedure, a total of 349 Class I supports installed since July of 1979 were inspected. Nonconformances were documented on 225 supports. Table III-1 is a tabulation of the types of of nonconformances reported on each system.

Since any deviation from the design drawings resulted in a nonconformance, many of the nonconformances were of a minor nature and did not affect the function of the support. One hundred forty-eight (148) supports fell into this category. They were determined by the District to be acceptable as is and required only drawing changes to reflect the as built condition. The District evaluated 77 nonconforming supports for their effect on system operability. Seventy-five (75) were determined to be acceptable as is, but were reworked to restore the supports to their design condition.

Only two nonconformances were determined to exceed code allowable limits. They involved missing frames at each of the supports located between two solenoid valves on the "A" and "B" RCS emergency vent piping. The frames are needed to restrain the piping in the horizontal plane and for uplift forces during a seismic event. For normal operating conditions, the lines were found not to be overstressed due to the missing frame. However, they did exceed code allowable stresses for the seismic design basis event. The frames have been installed per the original design.

B. Supplemental 79-14 Walkdown

In determining the scope for the inspection of pipe supports, Nuclear Engineering used I&E Bulletin 79-14 As Built Verification Program as the bench mark for the as built configuration of the plant. The District's management requested a review by Nuclear Engineering of the 79-14 walkdown scope compared to the scope defined by other utilities and the current definitions of safety related and Class I.

III APPLICABILITY TO PLANT-WIDE SYSTEMS (Continued)

B. (Continued)

The District interpreted the scope of the original 79-14 walkdown to apply to the safety feature or Emergency Core Cooling Systems (ECCS) described in FSAR Section 6 and its supporting systems, as well as parts of the Auxiliary Feedwater (AFW) System. Using information from the walkdown conducted at the Trojan plant, the seismic classification of equipment in Appendix 5B of the USAR, and Revision 3 of Regulatory Guide 1.29, Nuclear Engineering developed a list entitled "Supplemental 79-14 Walkdown." Table III-2 shows a comparison of the original 79-14 Walkdown and the Supplemental 79-14 Walkdown.

After reviewing the results of the comparison, the District's Management Safety Review Committee authorized the walkdown of the balance of 79-14 systems. Using the new procedure generation requirements in the District's new Nuclear Engineering Procedures, Nuclear Engineering developed the procedure, Engineering Report (ERPT)-M0001, Balance of IE Bulletin 79-14 Evaluation (Attachment V), based on the original 79-14 walkdown and Bechtel's generic 79-14 walkdown procedure.

The District reviewed the walkdown inspection criteria contained in ERPT-M0001 against tolerances given in 79-14 walkdown procedures for the Trojan, Pilgrim, Point Beach, and ANO-1 plants and found them to be comparable. In some cases, the District's procedures were more stringent. Walkdowns were performed by both District and contract engineers after they had been trained in the use of procedure ERPT-M0001 and in inspection techniques.

The supplemental 79-14 walkdown resulted in the visual inspection of 401 lines. Table III-3 shows the types and number of nonconformances identified. Each of the nonconformances was evaluated using conservative, simplified calculational methods for comparison to design allowable limits. Approximately 120 nonconformances were determined to require fixes to restore design margins in the original design. All other nonconformances were determined to be acceptable and required only drawing changes.

The District will complete all fixes using its standard inspection practices, as supplemented by a final verification walkdown by Nuclear Engineering, prior to restart. Final drawing changes and detailed analyses which incorporate the walkdown information are continuing.

III APPLICABILITY TO PLANT-WIDE SYSTEMS (Continued)

C. NSEB 85-01 Task Force Walkdown

In addition to the 79-14 and expanded scope 79-14 walkdowns, the District recently (May 30, 1985) concluded a walkdown of a randomly selected sample of supports within the support systems of heating, ventilation, and air conditioning (HVAC), piping, cable trays, and conduits in the Nuclear Services Electrical Building (NSEB), Auxiliary, and Diesel Generator Buildings. No walkdowns were performed in the Reactor Building due to ALARA considerations. The samples located in the other buildings were felt to be indicative of the status of the supports within the Reactor Building. This walkdown was performed in response to the findings of the NRC inspection 85-01 performed January 7-11, 1985. The NRC issued a notice of violation due to this inspection identifying discrepancies found in the as built configurations of HVAC supports in the NSEB with respect to design drawings.

As an immediate action taken in response to the above notice of violation, the District performed a walkdown of all visually accessible HVAC supports which were of the same type as those investigated by the NRC team in the NSEB. During the course of this inspection, the District determined that minor discrepancies (i.e., deviations which did not jeopardize the intended function of the support) did exist. The engineering evaluation of those discrepancies revealed no adverse effect on the safety of the plant. In addition, a root cause evaluation was performed. The results indicated that one temporary quality control inspector had been involved in all of the discrepant cases identified by the NRC.

However, in order to identify if any safety concerns, or programmatic problems existed, the Nuclear Engineering Department Manager assigned a multi-discipline, multi-departmental Task Force to conduct further investigations.

During their investigation, the Task Force discovered additional minor discrepancies and documented them on NCRs. Engineering evaluations of all discrepancies showed that sufficient design safety margins existed because no design allowables were exceeded, and therefore, no safety concerns. However, the Task Force identified nine areas of programmatic weakness and determined appropriate corrective actions. These are contained in Attachment VI, the Task Force report titled, "Systematic Support Investigation for Rancho Seco Nuclear Generating Station In Response to NRC Audit 85-01." The District has committed to complete these corrective actions by October 1, 1985.

III APPLICABILITY TO PLANT-WIDE SYSTEMS (Continued)

C. (Continued)

In light of the recent events involving the leak in the high point vent system, the District management extended the Task Force assignment to respond to the Licensee Event Report (LER) generated by that leak, to perform the root cause evaluation, to review LER 79-14 to determine what corrective actions were taken, and to identify possible further programmatic problems and corrective actions. In addition, the Manager of Nuclear Engineering assigned the Supervising Mechanical Engineer to the Task Force. Although only preliminary conclusions can be formed prior to a thorough root cause investigation, the Task Force will address issues such as:

- Drawing/Design Control Training
- Cognizant Engineer Assignments
- Training to Recognize Reportable and Nonconforming (AP.22) Situations
- Living Schedule Effectiveness - Work Load vs Human Error
- Construction Organization - Role of SMUD System Engineers/Field Engineers
- Craft Labor Accountability
- Effectiveness of QA Surveillances - Possible permanent multi-discipline Task Force to discover programmatic/design/construction/inspection weaknesses before they cause problems.
- Effectiveness of Past Training
- Relationship Between Startup Completion and Final Inspection

The District will implement all appropriate corrective actions identified by the Task Force prior to the Cycle 8 refueling outage.

III APPLICABILITY TO PLANT-WIDE SYSTEMS (Continued)

D. Independent QA Walkdowns

Investigative QA activities started immediately upon notification of 1" "B" side high point vent line problem. The QA Supervisor assigned a team consisting of two District QA Engineers to perform a surveillance inspection of the pipe failure. The following activities took place:

- Failed vent line was photographed and lines were mapped out, including supports.
- Drawings were pulled for "A" side and "B" side high point vent systems.
- "A" Side vent lines were walked down for as built conditions.
- "B" Side vent lines were walked down for as built conditions.
- ECN packages, original drawings, and material certifications were pulled from QA vault for review.

At this point, QA and Nuclear Engineering started working together to evaluate problems stemming from the above reviews. QA also decided to perform an independent walkdown. One QA team member attended the Nuclear Engineering training class for walkdowns and attended walkdown meetings. The other QA team member pulled drawings and assembled walkdown packages similar to Nuclear Engineering and made final preparation for the actual walkdown. Surveillance and audit reports have been asembled.

Two QA walkdown teams were assembled to perform independent verification of the Nuclear Engineering Post 79-14 walkdowns. Each team consists of one design/stress analysis engineer and one Quality Control Inspector drawn from contractors not involved in the original design and construction effort or the current Nuclear Engineering walkdowns.

Seven piping walkdown packages from seven different piping systems were selected at random to verify the post 79-14 walkdown by Nuclear Engineering. The walkdown packages are as follows:

System	Calculation No.
MSS	#201
FWS	#49B
BWS	#75
RCS	#98
PLS	#113
RCD	#120 & #55A
SIM	#45

III APPLICABILITY TO PLANT-WIDE SYSTEMS (Continued)

D. (Continued)

The QA activities which have been completed consist of:

- Walkdowns of piping systems, measurements taken of piping runs and supports.
- Preparation of sketches of as built drawings of pipe routing and supports.
- Evaluation of walkdown findings.
- Comparison of QA walkdown results with Nuclear Engineering results and the generation of NCRs for the differences.

A QA surveillance report will be issued for each system package on, or before, September 1, 1985.

E. Conclusions on Generic Applicability

The District has considered the generic implications of the high point vent failure on the plant-wide support systems. The District concluded that to ensure no further safety concerns exist, a 100% walkdown of affected safety related Class I pipe supports which were subject to computerized stress analysis would be performed. These walkdowns are described above in Sections III A and B. Upon review of NSEB 85-01 Task Force walkdown results for random samples of nonpiping support systems, the District concluded that the conduit, cable tray, and HVAC support systems are safe as designed and constructed. The bases for this conclusion are discussed in further detail below:

- The 85-01 walkdown randomly looked at 208 conduit, tray, and HVAC supports (containing an estimated 2,080 inspectable items) with all engineering evaluations showing that worst case seismic/static stresses within code allowable.
- Piping systems undergo constant dynamic forces such as in-service flow vibrations, high pressures, and high temperatures. Conduit, cable tray, and HVAC systems undergo only static dead loads except for the possible short-term seismic event. The crack of the high point vent line has been attributed to long-term, high cycle fatigue due to in-service loads. This is not a factor in the nonpiping support areas. (It is worthwhile to note that conservatisms do exist in the piping systems as designed. The vent failure occurred in a line that was not in conformance with the design.)
- The Stress-Iso metric transfer of information problem is inherent in piping systems only.
- The dummy spool piece problem is inherent in piping systems only.

E. (Continued)

- Electrical conduit supports have the following conservatisms in design:

  - Each support is a three-way support (longitudinally and in both transverse directions).

  - Support steel members governed by AISC and AISI Codes with applicable allowable stresses.

  - Concrete anchors have a minimum safety factor of four. Taken from the NRC approved Bechtel Design Guide, C2.40.

  - Welding procedures follow AWS and AISC Codes with applicable allowable stresses.

  - Unistrut fittings (connections) have minimum safety factor of three (3).

  - Peak seismic accelerations are used. (Conservative)

  - Conduit supports are calculated for maximum possible dead load which is conservative with respect to actual loads.

- In addition to the support and anchor items above, the electrical cable tray supports also have the following conservatisms in design:

  - Cable tray design dead load assumes a 50 lb per square foot loading which exceeds 100% cable fill. The Rancho Seco target cable fill is 40%.

  - Design is based on Bechtel Design Guide C2.7 (NRC approved). Accelerations are peak values using conservative damping values.

  - Designs are based upon maximum span lengths of 8' vertical, 24' transverse, and 48' longitudinal. Field lengths are often much less.

- HVAC supports have the following conservatisms in design:

  - Peak accelerations, or 1.5 times the acceleration corresponding to the fundamental frequency of the duct is used in design.

E. (Continued)

- - Concrete anchors have a minimum safety factor of four. Taken from the NRC approved Bechtel Design Guide C2.40.
  - Welding follows AWS and AISC Codes with applicable allowable stresses.
  - Support steel members governed by AISC and AISI Codes with applicable allowable stresses.
  - Ducts are conservatively supported at all dampers and changes in direction.
  - Main ducts are supported on each side of a "T" connection.
  - Many supports are designed to support concentrated loads which are conservative with respect to normal operating loads.

- Cracks in a piping system result in the loss of the systems contents. Cracks, yielding or other "failures" of the electrical/HVAC support systems are not as likely to result in loss of system function.

- On July 17, 1985, the chairman of the Task Force contacted Mr. Sam Swan, an Associate in the EQE, Inc. consulting firm of San Francisco, California. Mr. Swan has an extensive background in mechanical and nuclear engineering and has visited the sites of several destructive earthquakes. He has directed earthquake engineering projects for numerous United States and foreign utilities involving all safety-related mechanical and electrical systems. He is an expert on seismic qualification and testing and has pioneered the use of earthquake experience data for the evaluation of nuclear power plants. He holds degrees from the University of Arizona and Stanford University.

  The Seismic Qualification Utility Group (SQUG), of which SMUD has been an active member for over two years, selected EQE, Inc. to collect experience data from over 50 sites that have experienced major ground motion during nine actual earthquakes. EQE, Inc. has accumulated an extensive data base of all failures. It is also important to note that the NRC has also been actively involved in the SQUG effort during this same period of time, and supports the results of the experience data which will be used to help satisfy the requirements of the upcoming USI A-46.

  Most of the data base sites are standard commercial fossil plants, switchyards, and varieties of other facilities which are not designed to the rigorous standards of a nuclear power plant. Mr. Swan made the following observations:

III APPLICABILITY TO PLANT-WIDE SYSTEMS (Continued)

E. (Continued)

- EQE has collected extensive data on the performance of cable tray with ground motions up to and possibly exceeding 0.5 g. With miles of accumulated tray data, there was only one incidence of a failure. An outdoor cable tray was supported at the top of a high column and the column failed. Mr. Swan mentioned that this isolated incident was not representative of a nuclear power plant design. There were no other failures.

- From experience data on miles of conduit installations, there have been no observed failures. Mr. Swan also emphasized the fact that conduits do not need to maintain a pressure boundary.

- The data on HVAC supports has also resulted in only one isolated failure. The estimate ground motion at that particular site probably exceeded 0.5 g and Mr. Swan stated that the ducts were not designed to the same rigorous standards of a nuclear plant.

In summary, the reinspection of random samples of conduit, cable tray, and HVAC support systems during the NSEB 85-01 Task Force walkdown indicated <u>no</u> unacceptable discrepancies in those support systems. The conduit, cable tray, and HVAC support systems are subject to a different design routine which does <u>not</u> include a Stress-Isometric to construction drawing <u>transfer</u> as does piping support design. The service conditions of the nonpiping support systems are significantly less rigorous than that of piping systems (albeit the piping designs are conservative). The conduit, cable tray, and HVAC support designs are extremely conservative and the postulated failure of those supports is highly unlikely to result in loss of system function.

After thorough evaluation of the above points with respect to conduit, cable tray, and HVAC supports, the District is confident that those support systems are safe as designed and constructed.

TABLE III-1

POST IE BULLETIN 79-14

SUPPORT INSPECTION PROGRAM

TYPES OF NONCONFORMANCE

BY SYSTEM

SYSTEM / TYPES OF NONCONFORMANCE	NSW	WGS	MSS	DHS	FWS	RCS	HVS	TOTALS (BY TYPES)	FIXES
WELD ITEMS	0	0	4	1	13	23	63	104	12
SUPPORT LOCATION	0	0	0	1	0	6	66	73	0
GAPS TOO LARGE	0	0	0	1	2	3	56	62	50
MEMBER SIZE	0	2	0	0	3	10	26	41	2
ANCHOR BOLT SPACING	0	0	0	2	1	1	19	23	2
SUPPORT FUNCTION	0	0	0	0	1	5	10	16	2
BASEPLATE	0	0	0	0	2	4	7	13	0
ORIENTATION	0	0	0	0	0	2	1	3	2
PART TYPE	0	0	0	0	0	0	1	1	0
MISSING PARTS	0	0	0	0	0	10	0	10	10

COMPARISON OF ORIGINAL AND SUPPLEMENTAL
IE BULLETIN 79-14 WALKDOWNS

* CLASS I EQUIPMENT USAR APPENDIX 5b & REG GUIDE 1.29 Rev 3	ORIGINAL 79-14 WALKDOWN	SUPPLEMENTAL 79-14 WALKDOWN
1. REACTOR VESSEL & INTERNAL & CONTROL ROD DRIVES	NOT REQUIRED	NOT REQUIRED
2. RCS (SGs, PZR, RCPs) & PIPING, VENTS, DRAINS IN CONTAINMENT	PARTIAL	REMAINDER
3. CONTAINMENT PENETRATIONS TO FIRST ISOLATION VALVE	PARTIAL	REMAINDER
4. MS & FW PIPING UP TO STOP VALVES	PARTIAL	REMAINDER
5. ADVs & MSSVs	NO	YES
6. PORTIONS OF FUEL HANDLING EQUIPMENT	NO	NO
7. AFW & CST	PARTIAL	REMAINDER
8. DIESEL GENERATORS & FUEL SUPPLY	NO	YES
9. RB CRANE	NO	NO
10. ELECTRICAL CLASS I	NO	NO
11. NSRW (EXCEPT CHEM ADDITION & POND RECIRC)	PARTIAL	REMAINDER
12. NSCW (EXCEPT CHEM ADDITION & SURGE TANK)	PARTIAL	REMAINDER

COMPARISON OF ORIGINAL AND SUPPLEMENTAL
IE BULLETIN 79-14 WALKDOWNS

* CLASS I EQUIPMENT USAR APPENDIX 5b & REG GUIDE 1.29 Rev 3	ORIGINAL 79-14 WALKDOWN	SUPPLEMENTAL 79-14 WALKDOWN
13. RB SPRAY	PARTIAL	REMAINDER
14. EMERGENCY COOLING & AIR RECIRC	PARTIAL	YES
15. LPI	PARTIAL	REMAINDER
16. CORE FLOOD	PARTIAL	REMAINDER
17. HPI	PARTIAL	REMAINDER
18. BWST	NO	YES
19. PORTION RADWASTE TREATMENT SYSTEM	NO	NO
20. PORTION OF SFC SYSTEM	NO	YES
21. PLANT VENT	NO	YES
22. PLANT AIR CLASS I ACCUMULATORS	NO	YES
23. BORIC ACID ADDITION	NO	YES
24. CR/TSC & NSEB REFRIGERANT PIPING (1985 MODIFICATION ITEM)	NO	YES

* Includes large bore pipe and computer analyzed small bore pipe.

TABLE III-3

SUPPLEMENTAL IE BULLETIN 79-14 PIPING WALKDOWN PROGRAM

TYPES OF NONCONFORMANCES

TYPE OF NONCONFORMANCE	TOTALS * (BY TYPES)	FIXES** REQUIRED
Weld Items	58	16
Support Location Not Within Tolerance	105	2
Support Gaps Not Within Tolerance	103	16
Member Size Differs From Design	20	2
Anchor Bolt Spacing, Size, or Length	6	0
Support Function	97	22
Baseplate Discrepancies	16	0
Support or Valve Orientation	16	1
Part Type Not Correct	38	10
Missing Parts or Parts Present Which Were Unnecessary	32	15
Incomplete Design Details or Stress Isometric, or Drawing Not Available	78	5
Support Configuration Discrepancies	64	8
Pipe Routing and Valve Location Discrepancies	86	3
Others	44	9

* Preliminary Figures
** Estimates

IV CORRECTIVE ACTION PROGRAMS

The District is implementing both short and long term corrective actions in response to the high point vent pipe failure and the underlying configuration deviations. The short term corrective actions rectify the pipe failure and ensure that the plant is in a safe configuration. The long term actions are aimed at virtually all aspects of the District's nuclear operations to ensure that the plant remains in a safe configuration throughout its operating life.

A. Immediate Corrective Actions

To ensure that the plant is in a safe operating configuration, the Management Team has directed that the following corrective actions will be completed prior to restart of the plant:

- Completion of the walkdowns identified in Sections III, A, B, and C above;
- The repair of the "B" loop RCS high point vent piping and supports;
- Disposition of all Nonconformance Reports generated by the walkdowns;
- Completion of all resultant pipe support rework. This rework will be subject to the standard District inspection practices. In addition, the rework will be inspected by the walkdown teams.
- Suspension of the majority of Class I new construction modification work until the new District procedures are in place and training is complete. Any Class I work deemed to be essential will be subject to the District's standard inspection program and a separate verification by the Design Engineering Team.
- The District's two independent Quality Control groups have been combined and reassigned to the Quality Assurance Department. This will ensure independence of the quality control function.

B. QA/QC Reorganization

The District has spent considerable energy in the past year developing a more efficient and effective "Quality" operation. This effort included the restructure of the entire nuclear organization to make it more efficient and enhance the quality program.

One positive action was the merger of QA with QC (which was formerly divided between the Nuclear Engineering and Nuclear Operations departments) to form the Quality Department. This eliminated the real or perceived conflict of the "reviewing" organization being part of the "doing" organization. It emphasizes quality and not cost/schedule, and is a substantial move aimed at achieving the highest level of quality at Rancho Seco.

B. (Continued)

The District has taken action in the following areas to strengthen its Quality program:

- New Inspection Procedures

  New inspection procedures are being developed that are oriented solely toward inspection and are not encumbered with construction criteria. In the past, construction procedures were minimal and were used to govern the construction activity. Approximately 52 new procedures/standards are in the process of being written and the schedule calls for full implementation of the new inspection procedures/standards by January 1, 1986.

- February Training

  After the initial problems were uncovered regarding the weld hanger supports in the NSEB, training was given to the QC Inspectors. Even though the investigation established that one temporary inspector was responsible for the inspection oversight, all inspectors were trained. This training covered the following areas:

  - Specific duties of QC Inspectors
  - Interface with Field Engineers
  - Pertinent procedures
  - Responsibility for acceptance
  - Yellow-lining of drawings
  - NCRs
  - Prevailing codes
  - Compliance to design documents
  - Memos are not recognized as official documents
  - Validity of the inspection

- Inspectors - District Employees

  The District intends to substantially reduce the use of contract inspectors. By early 1986, the majority of all inspectors will be District employees. This will provide the desired control over the qualification, training, and specific assignments of these inspectors. It will permit the development of one inspection team trained to one set of instructions and one set of procedures. It will also permit the selection of individuals to supervise contract inspectors during an outage.

IV CORRECTIVE ACTION PROGRAMS (Continued)

B. (Continued)

- Consolidation of QA and QC

The advantage of merging QA and QC is that this consolidation provides an integrated approach for monitoring and evaluating the District's conformance to the Quality Assurance program. It permits more flexibility in making assignments and less duplication of effort. The collective expertise can be used to monitor construction and plant operation. Immediate feedback from the inspectors in the field as to the success or failure of the quality program is the desired goal. In certain cases, District inspectors can be sent to vendor shops to inspect equipment prior to its arrival at Rancho Seco.

- QA Engineers Dedicated to Support QC

Quality Assurance will dedicate engineers to support the QC function and will address concerns raised by the QC inspectors. This support will be more direct than in the past and will not be impacted by other priorities.

- QA Trending Program

Quality Assurance has become increasingly active in trending plant parameters and various programmatic indicators. This information is widely distributed and is reviewed in session by the Management Safety Review Committee (MSRC). This has resulted in high visibility of areas where either negative trends or unacceptable results are identified. For example, in the past, drawings have not being updated within an acceptable time frame. The identification of this trend resulted in positive corrective action being taken to reduce the backlog. The emphasis on trending has been recommended by NUMARC, INPO, and the NRC. The District recognizes the need for a good trending program and has dedicated the resources to achieve this goal.

- Training

One area that the District has identified as requiring more attention is training. Special training has been given to the Quality Assurance Lead Auditors by General Physics. This was a structured program complete with reading assignments, 20 hours of lecture, and finally, an examination with a required passing grade of 80%. The program was similar to that given to the Plant Review Committee (PRC)/MSRC, but was more extensive in the quality area. Other programs including simulator training are scheduled and/or are now being attended by QA personnel.

The QC personnel received special training in February with regard to configuration control. QCI No. 7, Quality Control Training Program, has been approved. This QCI outlines a very structured training program for the QC inspectors.

IV CORRECTIVE ACTION PROGRAMS (Continued)

B. (Continued)

- Quality Engineering

  In keeping with the expanded scope of quality activities, Quality Engineering will establish a program to monitor engineering design activities. Personnel requirements for this activity will be included in the 1986 budget and recruitment is now underway.

- Critical Self Appraisal Program

  The critical self appraisal program has been developed. Corrective action is appraised to determine whether the objective has been met or not. If not, the assessment must address a new or different approach. The new approach is then assessed after the corrective action program is completed. All this is done with high management visibility to ensure the problem has been correctly addressed.

C. Nuclear Engineering Procedure Program

During the past 10 months, the Nuclear Engineering Department has embarked upon an extensive program aimed at producing a fully integrated system of procedures and references for the design and construction of all modifications to the Rancho Seco Nuclear Generating Station. The Nuclear Engineering Procedures (NEP) Manual is designed:

- To meet the District commitment to the NRC and commitments for design control;

- To ensure that design and construction activites are performed uniformly, consistently, and efficiently; and

- To ensure that the continuity of design documents is maintained in accordance with the Rancho Seco Quality Assurance Program.

The Nuclear Engineering Procedures Manual is organized into the following major sections:

1000 Introduction
2000 SMUD Nuclear Organization Description
3000 Nuclear Engineering Procedures (Administration)
4000 Nuclear Engineering Procedures (Design, Procurement, Licensing)
5000 Nuclear Engineering Design Interface Documents
6000 Construction Engineering Procedures & Inspection Standards
7000 Nuclear Engineering Reference Library

There are many improvements in the new procedures. For example, NEP.4109, Configuration Control, now has requirements for design verification, pre-design walkdowns, construction review of DCNs, environmental qualification, and other procedural changes that will reduce the possibility of design errors.

IV CORRECTIVE ACTION PROGRAMS (Continued)

C. (Continued)

Nuclear Engineering has also recently added a training staff which will improve performance. Training is currently complete on NEP.4109, and is ongoing for the other sections.

The District has initiated the development of a Drawing Change Notice (DCN) preparation training session. This training has not been provided as formal class instruction to date as the decision was later revised to include the DCN preparation guidelines into a Nuclear Engineering Procedure (NEP.4112, scheduled for issue in September, 1985). In the interim, discussions with Nuclear Engineering's Supervising Engineers and Nuclear Operations document control supervision have been conducted.

In August, 1983, the District began a program to incorporate outstanding DCNs "in house." The purpose of the program is:

- To assume control of the backlog of outstanding DCNs and the impending volume of DCNs generated for the 1983 outage.
- To expeditiously and efficiently turn the work complete DCNs into distributed drawings that reflect current plant configuration.
- Provide access to technical personnel to resolve problems as expeditiously as possible.

In addition, the District has taken the following steps:

- Introduced tighter drawing/DCN controls;
- Increased staff to support drawing controls;
- Developed and issued procedures to improve overall document handling;
- Transferred responsibilities for drawing/DCN control to those departments most concerned with their effectiveness;
- Installing equipment to facilitate a shorter distribution time of updated drawings;
- Exploring methods to improve distribution of critical drawing updates;
- Improved the document handling interface with A-Es and vendors;
- Improved the supplier technical manual distribution/update process;
- Improved updates/additions to equipment lists;
- Eliminated the backlog of updates to its master drawing/equipment lists.

D. Construction Engineering/Inspection Standards Program

The initiation of procedures specifically for use in the construction of major plant modification, began in 1980 as the scope of work grew to such an extent that existing plant procedures originally intended for maintenance became inappropriate to control the expanded construction work. These construction procedures developed over the years as they were needed to control the various modifications. At the time, the District believed that the extensive modification work was not a long term effort. Therefore, no programmatically comprehensive approach was taken (as would be done for new plant construction). Those procedures, although singularly sufficient and technically correct, lack the consistency provided by a comprehensive programmatic approach. Consequently, these procedures did not provide consistent guidance, policy, or understanding of expectation. This lead to a lack of consistency in the quality of work effort and attention to detail not commensurate with achieving the desired excellence in quality construction.

To correct this situation, the District directed the preparation of a new coordinated, comprehensive set of detailed procedures to assure high quality construction and strong quality control. The new procedures and guidelines will present clear guidance, policy, and expectations.

This new set of procedures (Construction Engineering Procedures/ Inspection Standards) are a part of the new Nuclear Engineering Department Procedures, and are coordinated with the Engineering Design and Specifications sections. To date, Nuclear Engineering has prepared over 40 new construction procedures currently in the review process.

These new procedures comprehensively cover the following areas affecting the work of construction:

- Organization and responsibilities
- Preparation and control of the procedures and their associated documents
- Training and personnel
- Housekeeping
- Use of reports and work packages
- Safety for personnel and equipment
- Administrative work controls

D. (Continued)

- Technical procedures associated with each disciplines' work (such as concrete placement, raceway installation, fabrication and installation of piping).

These procedures address all aspects of the construction process and present a consistent and complete program to the user. Where there are existing plant procedures that are appropriate, they are referenced for use along with any additional construction policy requirements. The procedures use a standard format that includes Purpose, Scope, Responsibilities, Procedure, Materials, Installation and Inspection, Documentation, References, and a standardized listing of inspection items.

These new procedures stress attention to detail and accountability for quality by the construction engineer using procedural controls and approvals for all work. They require and define greater formalized involvement and responsibility throughout the work for the construction personnel covering:

- Design, construction, craft, and inspection personnel
- Walkdowns prior to work initiation and immediately before work closure
- The use of standardized work and inspection listings in detailed work packages
- The construction engineer's inspection and signoff for work quality in detail prior to inspection by Quality Control Inspectors.

The Nuclear Engineering Construction Department has increased training for engineers and craftsmen to assure common understanding and expectation, and provide for enforceable accountability for all construction personnel. These concepts are emphasized with increased training in procedures, special processes, quality control, and skill confirmation, stressing quality goals and personnel responsibilities, as well as technical and administrative knowledge.

The District presently expects that the procedures will be through final review by the end of September, with training running through mid November. Nuclear Engineering has committed to have the program fully operational as of January 1, 1986. Until the District has implemented these new procedures, the existing procedures (Construction Methods and Procedures/Engineering Inspection Instructions) will continue to be used. The District feels that the above schedule is acceptable since no major work is anticipated during this period and the existing procedures

D. (Continued)

are technically correct and are adequate during periods of low quantities of work. Any Quality Class I construction modification work done during this period, will require a separate 100% work check by Design Engineering and Quality Control immediately prior to turnover to Operations, in addition to the existing procedurally required checks and inspections. These additional interim inspections will provide adequate assurance that the work has been properly completed.

The District management has instructed Construction Engineers to check the quality of all work prior to requesting inspection from Quality Control, while emphasizing that they are held fully accountable for the quality of the work. Construction personnel and their supervision now place additional emphasis on attention to detail in both the inspection item documents and work execution such that all work is to be only of the highest quality.

E. Self Sufficiency Program

The District currently has a program for developing and implementing technical self sufficiency. It is the District's objective to depend less upon consultants and be able to do more of the technical work in house. The District has undertaken the following programs in each of the Nuclear Organization Departments:

- Nuclear Training Department

  The District has put together a plan to upgrade the Training Department and accredit the training program. In the past, the District has relied on outside consultants to provide training to District operating and maintenance personnel. The District is now implementing a program wherein all training personnel will be permanent staff members. This includes all instructors for the Senior Operator License training as well as maintenance personnel, technical personnel, health physics personnel, emergency preparedness personnel, and others.

  The District is presently writing a specification for a site-specific simulator. All operating personnel required to have a license will be trained on the new simulator. By having its own simulator, the District will also be self sufficient in running test cases on the plant response to various initiating incidents. The District would no longer have to rely on B&W to run simulator cases such as those run for the high point vent system (see Section II B).

E. (Continued)

- Nuclear Operations

The District has contracted with the Electric Boat Division of General Dynamics to review and recommend a staffing program for the Rancho Seco Chem-Rad Group. This program will involve a review of the District's present organization, comparing it to other utilities' health physics operations, and recommending the optimal reorganization and additions to the District's staff. The goal of this program is to man the chemistry, health physics, and radwaste control functions with permanent personnel, with the option of still using supplemental health physicist technicians during refueling outages.

Another of the District's self sufficiency programs is a review of the District's control of outage management, planning, and scheduling. This review is in the developmental stage, with a goal of obtaining proper permanent District staffing based on similar nuclear plants.

- Licensing

A year ago the District had only two engineers dedicated to NRC licensing work. Since that time the District added four new licensing engineers with one position vacant. The District believes that this has vastly improved communications with the NRC and has helped reduce dependency on outside consultants.

- Quality Assurance

The District is establishing the Quality Engineering Organization (QEO) within the QA Department. The QEO will be split into two groups. The first group will be Engineering Review and it will address the conformance of the Nuclear Engineering organization to the NEP procedures. The second group will concentrate on maintenance and operation, and their conformance to the maintenance procedures.

The Quality Control Organization, also in the QA Department, will be headed by a Quality Control Supervisor with plant experience. Currently, only 8 out of 32 inspectors are District employees. The goal is that all the others become District employees no later than the first quarter of 1986, except for necessary competent contract personnel to provide inspections during outage conditions.

Quality Assurance will acquire an engineer that has environmental qualification experience to ensure that all of the requirements of 10 CFR 50.49 will continue to be met.

E. (Continued)

During 1985, in keeping with its goals and objectives of setting up a surveillance program in the operational area, the District recruited a Senior Quality Assurance Engineer and three additional engineers: one having a background in Health Physics, one with Chemical Engineering background, and the third with a Senior Reactor Operators License. This team provides surveillance activities on the day-to-day operation of the plant.

The end result of this program will be the acquisition of the discipline/know-how permitting the District to have in house expertise in areas where it was not formerly available. These increases in technical self sufficiency will reduce the need for the District to rely on outside organizations to perform special audit/review services.

- Nuclear Engineering

The Nuclear Engineering Department (NED) presently has a staff made up of District personnel and personnel from several architect/engineering consulting firms. As of this date, there are 44 degreed District engineers in NED who direct the efforts of 230 consultant engineers, giving a ratio of 1 District Engineer per 5 consultants. The District Board of Directors has approved a staffing program within NED which would result in a ratio of at least 1 District engineer per 1 consultant engineer. The implementation schedule for adding an additional 71 engineers requires that 10 District engineers be hired in the remaining part of 1985; 30 engineers in 1986, and 31 engineers in 1987. The District has committed to evaluate this decision in mid-1986 and determine if a 1 to 1 ratio is a proper ratio of District personnel to consultants.

These additional engineers will be assigned to the engineering design and construction disciplines with the intent of replacing contractor personnel on a one for one basis with respect to both experience and skills.

IV CORRECTIVE ACTION PROGRAMS (Continued)

F. Root Cause Program

The District is establishing a six member Incident Analysis Group (IAG). The charter of the IAG will be defined in an Inter-Departmental Procedure which applies to all nuclear departments and is signed by the Assistant General Manager, Nuclear. The subsequent description is preliminary and may change somewhat as the program is implemented and experience is gained. The members of the IAG will have significant nuclear experience and will have expertise in the following disciplines:

- Mechanical
- Electrical
- Quality Assurance
- Radiation Control/Chemistry
- Operations

The purpose of the IAG will be to investigate significant incidents and determine the facts surrounding the incident, identify deficiencies, and implement corrective actions. By identifying all deficiencies and implementing thorough corrective actions, the root cause of the incident will be corrected.

The IAG will recommend a Recovery Plan for each incident that it investigates. This Plan will list each corrective action associated with the incident, the responsible person and the action due date. Each Recovery Plan will be reviewed and approved by the Management Review Team (MRT) before implementation. The MRT will consist, at a minimum, of the following members:

- Assistant General Manager, Nuclear (Chairman)
- Manager, Nuclear Engineering
- Manager, Nuclear Operations
- Manager, Quality Assurance
- Supervisor of Nuclear Licensing
- Supervisor of the IAG (Secretary)

All of the corrective actions on each Recovery Plan will be entered into the District's computerized commitment tracking system. The District Quality Assurance Department will perform a surveillance of each completed corrective action. The MRT will review the collective effectiveness of all corrective actions to ensure the root cause of the incident is addressed.

IV CORRECTIVE ACTION PROGRAMS (Continued)

G. Lessons Learned Program

The District will implement the Lessons Learned Program (LLP). A draft procedure has been developed and the subsequent description of this program is preliminary and may change somewhat as the program is implemented and experience is gained. The purpose of the LLP is to critique certain plant events within two days of the occurrence and to inform the appropriate plant employees within five working days of the event. The incidents examined by the LLP will be critiqued by a committee composed of the individuals directly involved in the occurrence. The committee will be chaired by an engineer from the Regulatory Compliance Group who will be professionally trained as a facilitator. The facilitator will encourage free discussion between those individuals participating in the critique and will ensure that the discussion remain productive and on a professional level.

The facilitator will prepare a critique report summarizing the cause, generic implications, and safety significance of the incident. The critique report will also address the lessons learned from the incident and will be distributed to those groups who will benefit from the lessons. The critique report will not identify any of the individuals involved in the event nor will it review incidents involving disciplinary action.

H. Consultant and Contractor Qualifications and Accountability

The District has prepared a draft (now in final approval process) program wherein the District's engineering supervisors review and approve the resume's of all proposed engineers sent to Rancho Seco to work on design and constuction modifications. The supervisors are accountable for the quality of the work for all personnel brought into the program. The District, in all of its consultant contracts explicitly states that: "SMUD shall have the right to interview, select, remove, or replace any consultant project personnel working on the Rancho Seco No.1 project." The District has, in the past, exercised this right and has removed and replaced engineering personnel who the District thought failed to meet its requirements.

The District has written a nuclear organization-wide statement defining accountability. The theme of this accountability statement is that individuals working at Rancho Seco will be held accountable for the results of their work efforts. The District is formally sending this accountability statement to our architect/engineer consultants and its maintenance contractor to ensure they understand the requirements for working at Rancho Seco.

V RECENT DISTRICT ACTIONS TO STRIVE FOR EXCELLENCE

A. Nuclear Service Cooling Water Pump Breaker Evaluation

The nuclear service cooling water (NSCW) pump P-482A motor feeder circuit breaker has tripped several times in the past. Following these trips, the breaker was successfully reenergized after testing and resetting the overload unit. When the breaker tripped on June 12, 1985, electrical maintenance retested the overload unit and thought it to be defective as indicated by setpoint drift. A spare overload unit from the warehouse was installed on the breaker and after setting and testing had been completed, the breaker was successfully placed in service on July 23, 1985. The District then shipped the subject overload unit to the manufacturer for further investigation. In addition, to ensure that there was no common mode failure of the overload units, the District satisfactorily tested (several breakers required resetting) all the breakers from both "A" and "B" buses (including the redundant counterpart of the 3A18 breaker on "B" bus).

During the investigatic : of the breaker trip, District personnel noticed that the NSCW pump motor was drawing a slightly higher full load current compared to the nameplate rating. The investigation revealed that an earlier impeller modification (made during plant startup in 1974) increased the pump brake horsepower causing a higher full load current. The higher current would not have caused the breaker to trip if the overload unit was not defective (or incorrectly set). The reason for this conclusion is that the breaker tripped when the motor current was well below the minimum tripping current based on ± 10% tolerance on the overload unit as specified by the manufacturer. However, for a larger margin between the highest expected NSCW pump motor current and the lowest expected trip point of the motor breaker, the new overload unit setting has been increased from 375 amps to 400 amps. The overload unit on the redundant counterpart of 3A18 breaker on "B" bus has also been reset to 400 amps, tested, and put into service.

The District reevaluated the qualified life of the NSCW pump motors based on the higher operating current and determined it to be 19 years. This information will be used for replacement/rewinding of the motors under the District's equipment qualification program.

On July 8, 1985, Nuclear Operations filed an Occurrence Description Report (AP.22) concerning the malfunction of the 480 volt breaker serving NSCW Pump P-482A. (This event involved a different circuit breaker than the one described above.) The malfunction involved a "delayed closure" (i.e., the breaker did not close immediately upon command).

An initial inspection and testing by the site electrical maintenance staff neither identified the cause nor duplicated the event. A manufacturer's (Brown-Boveri ITE) field engineer was also unable to find the cause or duplicate the event.

A. (Continued)

District management appointed a Root Cause of Failure Investigation Team on July 10, 1985. The team consists of three electrical engineers, one from each of the following departments: Nuclear Operations, Nuclear Engineering, and Quality Assurance. The investigation began immediately and is still in progress. The following paragraphs describe the actions taken to date:

- The team compiled a composite factual sequence of events surrounding the malfunction. The composite is based upon direct interviews of the four Control Room personnel involved in the attempt to close the breaker. Concurrently, a listing of all identical circuit breakers installed in the plant was generated. This listing has been expanded to include similar but newer models of the same breaker type and is available should the investigation indicate generic testing or inspection is appropriate.

- The team searched the plant history files and INPO NPRDS data base for occurrences of the same type and found no identical events. The team also compiled and reviewed for possible impact the maintenance history of the breaker and the other identical breakers.

- A Quality Control Inspector performed a walkdown of all identical Class I breakers. The walkdown verified the serial numbers and installed locations of each of the breakers. The inspection results were documented on a Maintenance Inspection Data Report. Minor data errors and omissions found in the plant Master Equipment List have been corrected.

- The team prepared a list of all possible causes, both mechanical and electrical and then evaluated each cause in light of the facts surrounding the malfunction. Those causes which could clearly be eliminated by the facts were so identified. The team identified those causes that could only be evaluated by a factory teardown and inspection. A Special Test Procedure (STP-945) was written to address the remaining possible causes. This test has been run with no failures.

- The District shipped the breaker to the Brown-Boveri factory in Columbia, South Carolina. The factory disassembled, inspected, and tested the breaker. A District Engineer and Quality Control Inspector monitored this activity. The inspection/tests did not identify any deficiencies.

Early in the course of this investigation, it became evident that a Class II breaker (purchased to an identical specification as the Class I breakers) had been placed in Class I service without final approval of the required documentation. Additionally, the District raised questions concerning the June 12, 1985, tripping of the original NSCW Pump P-482A breaker. District management directed that the scope of the investigation be increased to include a review of the programmatic controls and an indepth analysis of the original breaker trips.

A. (Continued)

Analysis of the original breaker trips revealed that the running overcurrent trip setpoints of both the A and B NSCW pump breakers were too low (see discussion above). The District prepared an Engineering Change Notice (ECN) and has changed the setpoints. The District performed a similar engineering analysis on the other identical Class I breakers verifying that their setpoints were correct. Testing was then performed to confirm that the breakers would not false trip under worst case current load.

The District fully expects this ongoing investigation effort to result in better programmatic controls and improved plant reliability.

B. Diesel Generator Circuit Evaluation

While performing surveillance testing on the Nuclear Service Bus automatic unloading/reloading circuit, Nuclear Operations discovered a design deficiency in the emergency diesel generator control circuit. The discovery indicated that if the diesel generator was in its 15 minute shutdown cycle and its associated nuclear service bus lost power, the generator output breaker would enter an open-close cycling mode. This cycling mode was the result of a "close" signal from the undervoltage/unloading circuit and a "trip" signal from the diesel shutdown circuit.

A minor change was made to the diesel control circuit to eliminate the design deficiency and prevent a recurrence of the cycling condition. The design change was formally documented and tested and does not alter the operation of the Nuclear Service Bus automatic unloading circuitry. A design deficiency in the original diesel control circuit caused the cycling condition. It was not the result of any design changes made by the District.

It should be noted that a Safety Features Actuation Signal would have prevented the cycling condition since it would defeat the diesel shutdown circuit. A Safety Features Actuation Signal in conjunction with a loss of offsite power would have allowed the diesel generator circuit breaker to close and remain closed. The Surveillance Program tests the diesels in modes in which they are required to operate and provides assurance that other deficiencies to not exist.

C. Reactor Trip Breaker Failure

In response to GE Service Advice 175-9.20, District sent all of their Reactor Trip Breakers (RTBs) to GE in Atlanta, Georgia for refurbishment of trip shaft bearings (seven RTBs). The RTBs were then sent to B&W in Lynchburg, Virginia for further testing and certification (10 CFR 21).

C. (Continued)

NRC letter dated December 6, 1984, identified 13 items that should be included in the periodic maintenance of the RTBs. District letter (RJR 85-231) dated 05/03/85, committed to the performance of the 13 recommended preventive maintenance items. 11 of the 13 items were performed by GE and/or B&W as part of the refurbishment/certification program. The final two elements were performed by the District after receiving the RTBs. During the performance of the last item (the post maintenance functional test) one of the RTBs failed to trip on undervoltage device (UVD) actuation.

In response to the RTB failure, the District conducted a thorough investigation. Technical representatives were called to the site from both B&W, Lynchburg, and GE, Atlanta. When the problem was identified as an out-of-tolerance mechanical adjustment of the UVD, the District undertook a program to reverify the work performed by GE and B&W. A Special Test Procedure (STP) was written to verify/adjust/confirm the 11 items previously certified by B&W/GE. This STP was run on the six RTBs eventually placed in service in our control rod drive system. A post maintenance function test was also performed prior to using the RTBs.

The RTB which failed its post maintenance function test has not been altered. It has been replaced with a spare RTB which has passed the STP and the post maintenance functional test. The failed RTB is secured and the District has received final recommendations from the B&W Owners' Group. The District is preparing a detailed procedure to continue the investigation of the failure.

The preventive maintenance procedure for the RTBs will be updated to include the 13 items prior to the next regularly scheduled preventive maintenance (PM) of the RTBs due this fall. Subsequent PMs will continue to verify the 13 items until such time as vendor recommendations and/or engineering judgement dictate even higher standards for the RTBs.

The solution to the RTB failure problem represents a strong "think and plan before doing" approach. By doing so, the District was able to diagnose the problem and take corrective action without erradicating the condition which led to the original failure. In taking this approach to plan the failure analysis, the District was able to exactly pinpoint the cause and preserve it, thereby benefiting the entire industry. Any further work on the failed unit will certainly proceed along the same lines, with collection and preservation of data being the highest priorities.

D. Rancho Seco Review of the Davis-Besse Indicent

Following the Davis-Besse auxiliary feedwater (AFW) event on June 9, 1985, the District began an immediate short-term review of the Rancho Seco AFW system and its susceptability to the problems that occurred at Davis-Besse. The initial review consisted of the following:

D. (Continued)

- A task force was formed on June 14, 1985, consisting of engineers from the electrical, mechanical, and I&C disciplines to assemble and analyze significant documentation pertaining to the AFW systems at both plants.

- In parallel with the above, the Supervising Site Technical Advisor was sent to Davis-Besse to obtain additional information concerning the AFW system design and events of June 9, 1985.

- Information and conclusions obtained from the two actions above were presented to the PRC and MSRC.

The general conclusion of the short-term investigation, noted above, was that sufficient differences exist between Rancho Seco and Davis-Besse, to preclude the Davis-Besse scenarios from occurring at Rancho Seco. During this investigation, a number of less critical items came to light which require additional evaluation. Consequently a long term task force was formed consisting of members of Nuclear Operations, QA, Licensing, and Nuclear Engineering to address these issues. The general scope of the task force work is to evaluate all items which might result in the loss of the steam generator as a heat sink. The items, known to date, have been categorized as follows:

- Main Feedwater System Review

  Industry and District efforts currently in progress to upgrade main feedwater reliability will be identified and the system will be reviewed to identify possible changes to increase the main feedwater system reliability to limits challenges to safety systems.

- EFIC Review

  The EFIC system, due to be functional in the 1988 outage, will be reviewed to assure that in solving numerous problems, no new vulnerabilities are introduced.

- AFW System Characteristics

  This portion of the study will determine additional component and system operating characteristics, presently unavailable, so that a better understanding of potential problems in the AFW system, both pre and post-EFIC.

D. (Continued)

- Design Items

  During the short term review, various members of the task force, PRC, and/or MSRC proposed a number of potential design improvements. These are generally intended to remove from Rancho Seco design problems encountered at Davis-Besse.

- Human Factors Items

  In the short term evaluation, the task force conducted a thorough review of Human Engineering Observations (HEOs) identified in the conduct of the Rancho Seco Control Room Design Review (CRDR) (final report due 12/31/85). The conclusion of this review was that significant Human Factors Engineering (HFE) discrepancies do not exist at Rancho Seco. However, a number of relatively minor items surfaced which merit further review.

- Miscellaneous Items

  The task force will investigate a few additional items relating to maintenance, testing, procedures, etc., which do not conveniently fit within the other categories.

Since the effort required to perform this long term analysis is significant, estimated at approximately 1,000 manhours, the project will be controlled by the Living Schedule concept.

E. Equipment Qualification Program

The District's Environmental Qualification Program to meet the requiremnts of 10 CFR 50.49 reflects the District's commitment to strive for excellence. An Environmental Qualification audit was conducted by the NRC from May 20 to May 24, 1985. This audit reviewed:

- Master Equipment List Development
- Procedures
- Harsh/Mild Definition
- Resolution of IE Notices
- Qualification Files
- Maintenance Procedures

E. (Continued)

The NRC's comments at the exit interviews indicated that SMUD was in compliance with 10 CFR 50.49 and had a well thought-out and strong Environmental Qualification program. Only one issue was left unresolved by the end of the audit. The District provided additional information to the NRC within 45 days of the audit addressing this issue.

The District has hired a Senior Engineer to work in the Environmental Qualification group to ensure that the District remains in compliance. At least two additional people will be hired for this group within the next twelve months. In the interim, the District is committed to maintaining the Environmental Qualification Program with a combination of SMUD and consultant engineers.

F. Appendix R Program

In striving for excellence, the District included in its Appendix R Program the following elements of reassessment of past practice or actions:

- The District has reviewed all NRC commitments since 1975 relating to Fire Protection and documented by plant walkdowns and/or reviewing procedures that the commitments have been implemented.

- A complete review was made of all fire doors with Underwriter's Laboratory (UL). The District now is working to correct all fire door problems identified by UL.

- The District performed a complete walkdown of every fire barrier penetration. The barriers were analyzed and are being replaced if found defective.

- The Appendix R program included a review of all NFPA codes applicable to Rancho Seco and will correct any deficiencies in meeting the code.

The District is verifying and documenting what was done and, in a timely fashion, is eliminating any deficiencies found. In the interim, the District is implementing compensatory measures which will remain in effect until all of the above items are completely addressed. As a long term corrective action, the District will issue a fire protection program manual.

G. Silicon Bronze Bolt Failure Investigation

Several silicon bronze bolts were found to have failed in-service on the 3A nuclear service bus. An investigation was initiated to determine the scope and cause of the failures.

G. (Continued)

The District performed an inspection of all silicon bronze bolts in both the affected switchgear 3A bus and switchgear 3B bus as well as numerous bolts throughout the plant. No additional failures were found.

The investigation revealed no record of similar failures at Rancho Seco. In addition, the manufacturer knew of no similar failure of its product.

Metallurgical examinations of the failed bolts showed a brittle, intergranular failure. The failure was most likely environmentally caused and was similar to integranular stress corrosion cracking in that the failed bolts exhibited a poor grain structure in the vicinity of the failure. The failure of accelerated stress corrosion testing of similar bolts with good grain structure to cause cracking further supports this conclusion.

The bolt failure appears isolated to the 3A bus since the investigation revealed no other failures. The environmental conditions which may have contributed to the bolt failures have been eliminated.

The Root Cause Evaluation Task Force has directed disassembly of the bus tie bars for testing and will recommend long term corrective actions.

H. New Steam Generator Tube Inspection Program

Rancho Seco experienced three steam generator tube leaks in 1984. These tube leaks limited plant operation to very short periods during the latter half of the year. The District also found recurring evidence of a very small tube leak in the "B" Steam Generator.

In order to minimize steam generator tube leaks in the future, the District has implemented an aggressive Eddy Current Testing program. This program uses the most up-to-date techniques, such as:

- Multi-frequency/Multi-parameter analysis
- Digital data acquisition and analysis
- Multi-coil (8 x 1) probes
- High fill-factor (.540") probes.

The program includes a 100% inspection of all special interest areas in the generator with an independent second party analyzing all data.

H. (Continued)

The District's program requires the inspection of six times the minimum number of tubes that are required to be inspected by the Standard Technical Specifications. Further comparison reveals that the District's program requires two and one half times the inspection rate recommended by EPRI in their guidelines for B&W steam generators. Additionally, extremely high risk tubes are examined using both "bobbin-coil" probes and multi-coil (8 x 1) surface riding probes. High fill-factor (0.540") bobbin-coil probes were used in repeated examinations of selected tubes with indications which required additional evaluation.

Tubes to be plugged as a result of Eddy Current indications are identified using several approaches including: the Technical Specification plugging limit of 40%; and crack-like indications in high risk tubes. Decisions to plug tubes are made with an acute appreciation for the possibility of flaw growth, as well as the phenomenon of tube support plate wear. The decision process may result in the plugging of more tubes than is absolutely necessary. However, removable plugs are now used and sleeving may be possible in the future.

The District has taken additional steps to eliminate steam generator tube leaks. These include:

- The District has brought its chemistry guidelines into conformance with EPRI recommendations;
- Operating procedures now include "hot soaking" of the steam generators;
- The District has formed a "Steam Generator Health Committee" not only to eliminate tube leaks, but also to improve steam generator performance; and
- The District has obtained tube samples that are currently undergoing laboratory examinations.

I. Nuclear Fuel Handling Betterment Program

The District's management has recognized that the fuel handling activities at Rancho Seco could be greatly improved. In response to this recognition, the District's management established a steering committee consisting of experienced engineers and operations personnel to ensure that the Rancho Seco fuel handling activites are improved.

I. (Continued)

To date, many changes have taken place, with additional changes planned, that will improve the fuel handling operations. These changes are broad in nature, affecting not only hardware, but operating practices as well. By implementing the changes, the District will improve the fuel handling system reliability and operability leading to shortened fuel handling operations, improved plant availability and reduced operator exposures.

The reliability of the fuel handling system will be improved by the following actions:

- Revisions to the preventive maintenance procedures and maintenance instructions are being made to incorporate more detail and direction. Also, various aspects of performing maintenance tasks are being evaluated.
- The fuel handling system is being modified to incorporate an above water drive system of District design.
- The spare parts inventory will be improved. A review is under way to determine which parts to keep in inventory.

The availability of the plant will increase due to increased fuel handling system reliability, as discussed above, and improved system operability. The following equipment changes have been made to improve the system operability.

- An improved control rod assembly mast has been installed on the main fuel bridge to eliminate the need to perform manual control rod shuffles.
- An inching system will be installed to allow the operators to make "micro" movements of the equipment in the core.

Modifications are also being made to improve the serviceability of the equipment such as relocation of equipment to areas of lower exposure.

The refueling environment is being improved by the installation of new lighting and upgraded housekeeping practices.

The changes within the fuel handling system and in the operating practices will lead to improved system reliability and plant availability. In addition, the changes will reduce operator exposures and lead to a reduction of the District's dependence upon contractor personnel.

J. Meteorological Program

The District, over the past 18 months, has had a series of minor problems with the meteorological program. District management formed a Task Force to evaluate the program and recommend corrective actions. At an NRC exit interview early in 1985, Region V inspectors discussed the meteorological program weaknesses with the District management. Immediately following this exit interview, plant management established a Task Force comprised of the Supervisor of Emergency Preparedness as Chairman, the Nuclear Electrical Instrumentation and Control Superintendent, the Supervisor of Nuclear Fuels Management and Health Physics, the Supervisor of I&C Engineering, and a Quality Assurance department representative. This multi-disciplinary Task Force performed an indepth study of the meteorological program history and current status. The Task Force generated a report which had numerous recommendations, all of which were concurred in by the management team. Some of the more significant recommendations and corrective actions scheduled for implementation are as follows:

- Designate a Meteorological Program Coordinator.
- Obtain the services of a meteorological consultant on a contractual basis for periodic review of data sets and system performance in Quality Assurance audits.
- Reconfigure Channel A and Channel B displays or printouts such that Channel B instrumentation should be considered as hardware backup for Channel A and the availability of offsite meteorological data should be considered as system backup.
- Perform data reduction/manipulation using an approved procedure.
- Perform periodic QA audits.
- Install precipitation instrumentation.
- Correct the new MIDAS program averaging process to meet the Regulatory Guide 1.23 criteria for averaging data points.

The District firmly believes that it has strengthened its meteorological program as evidenced by the formation of the Task Force as well as the appointment of a Meteorological Program Coordinator.

K. Surveillance Program Evaluation

The District believes that the Surveillance Program is an effective tool to periodically assure that safety systems function as required by the designer in the original analysis. To ensure that the District's program is performing this function, the District's management directed an evaluation of the program.

K. (Continued)

The following major systems were chosen as a sample for verification:

- Auxiliary Feedwater System
- High Pressure Injection System
- Decay Heat/Low Pressure Injection System
- Reactor Protection System

Discipline engineers within the District's Nuclear Engineering Department reviewed design documents for the selected systems and produced a list of components and their associated safety functions. The plant's Technical Support staff then searched the Surveillance Program to determine whether or not these functions were periodically tested.

The District found that the Surveillance program adequately demonstrated the functions identified by the discipline engineers which included those specified in Section 4.0 of the Rancho Seco Technical Specifications. The District has therefore concluded that the Surveillance Program effectively assures, on a periodic basis, the Rancho Seco safety functions.

While the District has determined that the Surveillance Program verifies the Rancho Seco safety functions, in keeping with the District's commitment to strive for excellence, the program is undergoing improvements in the following areas:

- The program is generating a cross reference between source documents and the Surveillance Procedures. (Source documents are defined as the Rancho Seco Technical Specifications and letters of commitment to the NRC.)
- The District is writing a Surveillance Procedure "writer's guide" (draft currently under review) to give detailed information on the format, content, and style expected for all procedures. INPO documents are being used as an input to the guide and the existing procedures will then be evaluated aginst the guide and rewritten as necessary.
- The District is also preparing a Surveillance Procedure "biannual review guide." This document will lead an engineer through the reevaluation of procedures that occurs every two years. A key requirement will be that the engineer actually observe the execution of the procedure by the technician or operator.
- The District is planning a Surveillance Procedure results review guide to lead the reviewing engineer through this process of reviewing test results and assure attention to detail.

K. (Continued)

- The District is computerizing the scheduling of surveillances. This action is designed to obviate potential scheduling errors associated with manual scheduling.

L. Rancho Seco Trending Program

Rancho Seco Nuclear Operations personnel currently perform trending on a variety of equipment and parameters using data taken from the Surveillance Program to trend and ascertain the cause of failures or other problems. The program trends selected parameters from routinely acquired plant performance data.

The Preventive Maintenance Program obtains data for use in trending. Special circumstances, such as failures or unique concerns, prompts trending of data taken routinely during plant operation.

Currently, many Nuclear Operations divisions individually perform trend analysis. Although these analyses are not standardized, they do have the advantage of letting the individual divisions tailor their trending to their particular needs. To increase the effectiveness of these individual analyses, the District is developing a comprehensive program to improve the trending practices at Rancho Seco.

The District's program will use a part of its guidance, the INPO Good Practice on Trend Analysis which includes trending of:

- Surveillance and instrument setpoints,
- As found and as left data, and
- Chemistry data.

In addition to the above, the program will also trend:

- Preventive maintenance data,
- Calibration data on non safety related equipment,
- Failure modes and effects,
- Performance efficiency data, and
- Consumable supplies usage and maintenance costs.

The Trending Program will serve as a "clearing house" for the Rancho Seco trending activities ensuring requested parameters are properly trended. The program will add consistency and expertise to trending activities at Rancho Seco.

VI SPECIFIC RESTART REQUIREMENTS

In line with the District's commitment to strive for excellence, the District's management has directed that the items discussed below be completed prior to restart of the plant. Many of these items are over and above those required to ensure that the plant is in a safe configuration.

- All NCRs generated during the walkdowns of Sections II A and B will be dispositioned.

- The District will complete all rework required by the above NCR dispositions.

In addition, the District has completed the following actions:

- Evaluation of the cracked "B" loop high point vent pipe and piping system;

- Determination of the applicability of the identified concerns on the plant-wide systems including completion of:

  - The post 79-14 walkdown;

  - The supplemental 79-14 walkdown;

  - The evaluation of NRC Audit 85-01 walkdown results with respect to conduit, cable tray, and HVAC supports;

- Repair of the "B" loop high point vent piping and supports;

- Documentation of nonconformances identified during walkdowns on NCRs, and

- Repair and testing of the nuclear service cooling water pump breaker.

VII CONCLUSIONS

The District is committed to strive for excellence in the operation of the Rancho Seco Nuclear Generating Station. This commitment is backed up by a management attitude that fosters an environment in which continually increasing standards and levels of quality are achieved.

The changes taking place within the District's nuclear program are evidence that the District management is successfully communicating its commitment and attitude throughout the Nuclear Organization. Many of these changes affect virtually all aspects of the District's nuclear program. Previous sections of this report discuss these changes and are summarized below:

- Management consultants reviewed the District's Nuclear Organization to provide suggestions as to organizational changes to strengthen the nuclear operation. One aspect of this reorganization was the combination of previously separate Quality Control groups into a single division and the placement of that division into the QA Department. This move should strengthen the quality control function. The District has also begun to implement other reorganization recommendations.

- The District has been in the process of performing self-critical reviews of its operating practices and making programmatic improvements. This process is an ongoing effort.

- The District has embarked on a program to strengthen its technical self sufficiency. This program has an objective of reducing the District's dependence upon consultants and will be implemented in virtually all of the nuclear departments.

- A large majority of the technical staff is located at the plant site to facilitate design implementation and improve communications.

- The District's design and construction procedures are being formalized and revised to ensure attention to detail and that proper design and construction practices are used.

- The District has a root cause investigation program to ensure that lessons are learned from problems and to minimize the possibility of recurrence. This program is in the process of being formalized and expanded.

In addition, the District has completed the following actions which exemplify its commitment to excellence:

- Verification that the Surveillance Program demonstrates the plant safety functions;

- Diesel Generator circuit modification and testing;

- Implementation of the Equipment Qualification Program; and

- Implementation of improved steam generator tube inspection criteria.

VII CONCLUSIONS (Continued)

The changes and programs discussed above are the result of the District's commitment to excellence and the management attitude that backs up that commitment. The changes are indicative of a continuing process that not only ensures that Rancho Seco is operated safely, but that plant operations improve throughout the plant's lifetime.