U. S. NUCLEAR REGULATORY COMMISSION

REGION V

Report No. 50-312/85-19

Docket No. 50-312 License No. DPR-54

Licensee: Sacramento Municipal Utility District
P. O. Box 15830
Sacramento, California 95813

Facility Name: Rancho Seco Unit 1

Inspection at: Herald, California (Rancho Seco Site)

Inspection conducted: June 23 - August 9, 1985

Inspectors:

J. H. Eckhardt, Senior Resident Inspector	8-22-85 Date Signed
G. P. Perez, Resident Inspector	8-22-85 Date Signed
J. L. Crews, Senior Reactor Engineer	8-22-85 Date Signed
A. J. D'Angelo, Resident Inspector, San Onofre	8-22-85 Date Signed
H. J. Wong, Office of Inspection and Enforcement	8-22-85 Date Signed

Approved By:

L. F. Miller Jr., Chief, Reactor Projects, Section 2	8-22-85 Date Signed

Summary:

Inspection between June 23 - August 9, 1985 (Report 50-312/85-19)

Areas Inspected: This special inspection evaluated the causes of the B hot leg high point vent leak, and included the areas of engineering and design control, construction, quality control inspections, and quality assurance oversight. This inspection involved 298 hours by five inspectors.

Results: Enforcement actions relative to the issues discussed in this report will be the subject of separate correspondence.

DETAILS

1. Persons Contacted

   a. Licensee Personnel

      R. Rodriguez, Assistant General Manager, Nuclear
      *P. Oubre', Manager, Nuclear Operations
      *G. Coward, Plant Superintendent
      L. Keilman, Manager, Nuclear Engineering
      L. Schwieger, Manager, Quality Assurance
      *J. Jewett, Site QA Supervisor
      *J. McColligan, Nuclear Project Engineer
      *J. Field, Technical Support Superintendent
      *D. Abbott, Supervising Mechanical Engineer
      *S. Crunk, Regulatory Compliance Engineer
      J. Barker, QA Engineer

   b. Bechtel Personnel

      *T. Kahn, Group Supervisor, Mechanical

   In addition to the above personnel, the inspectors contacted other supervisors, operators, engineers, and inspectors during this inspection.

   *Attended the exit meeting on August 9, 1985.

2. Description of the Event

   On June 23, 1985, the reactor was in a hot shutdown condition (reactor coolant system at 532°F and 2145 psig). At 0405 hours, with the reactor approaching criticality, the control room operators detected a reactor coolant system (RCS) leak. The control room operators estimated the leakage to be 17 gallons per minute. Using the reactor building camera, the operators were able to detect a steam leak in the vicinity of the "B" once through steam generator (OTSG). The control room operators then initiated a plant cooldown, and declared an unusual event due to a RCS leakage rate of greater than 10 gallons per minute. At 0735 hours a licensee inspection team entered the reactor building in an attempt to isolate the leak. The leak was verified to be on a one inch line in the "B" high point vent system and not isolable. The plant cooldown and depressurization occurred in an orderly manner. The plant reached a cold shutdown condition at 1838 hours on June 23, 1985, and secured from the unusual event. By the time the RCS was depressurized, approximately 16,000 gallons of the coolant leaked into the reactor building. There was no observable increase in the reactor building surface or airborne contamination due to the coolant leakage.

   The source of the leak was found to be a crack located on a segment of piping that connects the "B" OTSG nozzle to the high point vent system and the nitrogen supply system for the RCS. The crack was a 120 degree through wall opening azimuthally about the pipe wall. The pipe was a one inch diameter schedule 160 line. The licensee's investigation

subsequently identified that the "B" high point vent system had deficiencies in the as-built pipe support configuration.

3. Background Information

The original construction of the Rancho Seco Nuclear Power Plant included two vent lines connected to the top portion of the RCS hot leg piping. The vent line on the "B" side included a one inch diameter quality Class 1, seismic Category 1 nuclear piping (Class 11N1) traveling from the hot leg steam generator nozzle to the first isolation valve (RCS-017). The section of the piping from the first isolation valve to the second isolation valve (RCS-011) was classified as quality Class 2, seismic Category 2 nuclear piping (Class 22N2). The remaining portion of the nitrogen supply piping was classified quality Class 3, seismic Category 3 (Class 33).

The licensee dispositioned a nonconformance report (NCR S-046) in October 1974, which called for the removal of a portion of the Class 33 piping in the nitrogen supply system. A removable spool piece was to replace that portion of piping and be removed during plant operation to prevent the potential contamination of the nitrogen supply system.

The inspectors were not able to locate any modification package for this change. Therefore, it appears that the licensee did not perform the proper stress calculations to assess the effect of the modification on the Class 11N1 portion of the system.

In response to a requirement of the NUREG-0737 "Clarification of TMI Action Item Requirements", the licensee initiated a design for reactor coolant system vents. The purpose of the vents was to remove noncondensable gases from the RCS which might inhibit core cooling during natural circulation. A contract was issued to Bechtel Power Corporation to design three vent systems; two were for the "A" and "B" sides of the RCS hot legs and the other for a vent system to connect to the pressurizer.

Pipe stress calculation No. 316 and the accompanying isometric sketch dated September 26, 1981, (Rev. 1 to Bechtel Job No. 12334) for the "B" side high point vent and nitrogen supply system identified the following changes to the existing piping:

a. "Spring hangers at points 295 and 345 replaced with rigid supports." This was to modify an existing guide hanger assembly described in drawing M-485 sheet 1-329.

b. "Point 247 connected to 248." This change was also identified in Note 1 on the isometric as..."Field to weld rigid member between two flanges." This was to be placed in the nitrogen supply piping between points 247 and 248 to support the piping with the spool piece removed during plant operation.

c. "Points 437 and 438 connected together by 1½"x1½"x¼" angle." This change was also identified in Note 2 on the isometric as..."Field to

install $1\frac{1}{2}$"x$1\frac{1}{2}$"x$\frac{1}{4}$" L, 12" long data point 437 and 438 with two U-bolts as shown."

d. "U-bolts at points 326 and 290 to be tight on pipe." This was for an existing spring hanger assembly described in drawing M-485 sheet 1-330.

e. "Rigid Z restraint added at 290 and at 295." This was to add horizontal restraints to the assemblies identified in items a and b above.

Furthermore, a Bechtel project engineer sent a letter to the licensee on October 7, 1981, discussing the changes that were necessary to modify the existing "B" vent piping as noted in a through d above.

The licensee installed the solenoid operated high point vent modification required by NUREG-0737 during the 1983 refueling outage. The work was described and controlled by engineering change notice ECN-A-2934. The existing vent and nitrogen supply piping support additions and modifications described in a through d above were included in changes to drawings M-485 sheets 1-329 and 1-330 and in the stress calculation isometric sketch and were part of the drawing package of ECN-A-2934. However, these support additions and modifications were not included or described in the actual ECN and were not installed. The quality control inspection conducted after completion of ECN-A-2934 did not identify that the additions and modifications were not installed.

4. Licensee Investigation

As a result of the June 23, 1985, pipe crack in the RCS, the licensee's safety review committees, the Plant Review Committee (PSRC) and the Management Safety Review Committee (MSRC), developed an extensive investigation. In scope, the investigation was to determine possible cause(s) of the crack, the potential generic implications, and the corrective actions required with respect to the licensee's findings.

In response to the identification of the configuration difference between the "B" high point vent line and the design, the licensee initiated a detailed walkdown of all safety-related piping modifications made since the as-built verification required by NRC Bulletin 79-14, "Seismic Analyses for As-Built Safety-related Piping System" (IEB 79-14) (the "Post 79-14 Walkdown").

a. Post 79-14 Walkdown

The post 79-14 walkdown included all safety-related piping modifications installed since the as-built verification required by IE Bulletin 79-14. To ensure thorough walkdowns, a detailed inspection procedure was written and the walkdown crews were trained. The walkdown packages included both the stress analysis calculations and construction drawings.

This effort included 349 safety-related piping supports installed since the original IEB 79-14 effort and resulted in nonconformances

identified in 225 supports. Of these nonconformances, 148 were determined by the licensee to be of a minor nature so as not to affect the function of the support. They were determined to be acceptable as is and only drawing changes were made to reflect the as-built condition.

The remaining 77 nonconforming supports were evaluated by the licensee. Of those, 75 were reworked to meet the original design condition although they were determined to not effect system operability.

The remaining two supports were determined to exceed the code allowable limits for their related systems for a seismic design basis event. These supports were between the two solenoid valves on the A and B RCS emergency vent piping and involved missing frames needed for both horizontal and vertical restraint during a seismic event. The licensee has subsequently installed these two supports to meet the original design.

b. Supplemental 79-14 Walkdown

Based on reviews of the scope of the original 79-14 walkdown, the licensee determined that their original walkdown should be expanded to include all safety-related systems, and not just those emergency systems as originally performed. To perform the balance of 79-14 walkdowns (referred to as the supplemental 79-14 walkdown), the licensee developed a walkdown inspection procedure. The walkdown teams consisted of SMUD and contract engineers with expertise in the pipe design area who were trained to the walkdown procedure.

The supplemental 79-14 walkdown included visual inspection of 401 lines consisting of over 1800 components (supports and valves). Approximately 25 percent of the components were identified as having nonconformances. These were documented on 107 nonconformance reports. Many of the nonconformances were determined to be acceptable and only drawing changes were made to reflect the as-built condition.

Approximately 255 supports were determined to exceed the code allowable limits for supports for a seismic design basis event and require fixes to meet the design requirements for supports. The problems noted were of the same types as identified in the post 79-14 walkdowns and included items such as weld items, out of tolerance support locations, out of tolerance gaps, configuration discrepancies, etc. It is noted that the licensee did not identify any piping that exceeded code allowable limits even though the supports exceeded code allowable limits.

The licensee also performed an investigation of the pipe's failure mode, which included an elaborate metallurgical study of the pipe segment with the crack, and an analytical comparison of the calculated stresses needed to produce the crack versus the stresses calculated from the as found condition. The licensee also performed

a plant transient analysis on the possible consequences of a total severance of the one inch diameter pipe.

The quality assurance organization performed three special audits: an investigation of the cause of the "B" high point vent line break; an audit of the adequacy of the post 79-14 walkdowns; and an assessment of the supplemental 79-14 walkdowns.

5. NRC Evaluation

The inspectors reviewed the licensee's evaluation of the failure mode for the vent system piping. The licensee's evaluation was based upon an investigation conducted by General Electric Co., Nuclear Energy Engineer's Division (GE), for their determination of the cause of the cracking.

The GE investigation consisted of optical and electron scanning microscopy, metallography, hardness measurements, elemental analysis, and gamma scanning. The results from the General Electric Co.'s tests concluded that the through wall circumferential transgranular crack was most likely caused by high cyclic fatigue. The metallugical evaluation appeared to be complete and comprehensive.

The inspectors also requested the licensee consider a mechanical vibration calculation of the as-found condition of the "B" high point vent system to verify that the configuration did have the potential to generate the stresses needed for the pipe failure. The inspectors reviewed the results of the calculation, performed by the licensee, and agreed with the determination that the piping system was sufficiently unsupported to create the cyclic load required for the pipe crack.

Based on the recent piping failure of the RCS vent line, the licensee's actions for IE Bulletin 79-14 were reviewed. This IE Bulletin (issued on July 2, 1979 with subsequent revisions and supplements) requested verification that the seismic analysis reflect the actual, as-installed configuration of safety-related piping systems. The Bulletin had been previously reviewed in Inspection Report Nos. 50-312/80-04, -09, -24, and -31.

The areas reviewed as followup to the licensee's activities for IE Bulletin 79-14 were: coverage of systems by the licensee, followup to selected LER's, review of the re-analyses of piping systems based on IEB 79-14 walkdowns, review of QA audits concerning the bulletin.

In addition, the licensee's QA surveillance of the engineering walkdowns, dispositioning of nonconformances, and pipe failure mode analyses were reviewed.

a. Guidance was provided in the IEB 79-14 as to which safety-related piping systems were to be considered under the bulletin. Safety-related piping was to include Seismic Category 1 systems as defined in Regulatory Guide 1.29 or the FSAR, and also Seismic Category 1 piping regardless of size if it had been dynamically analyzed by computer.

The SMUD engineer who was cognizant of the previous IEB 79-14 actions stated that the philosophy that was used in the selection of systems was that only those portions of systems were analyzed which were engineered safety features systems (i.e., high pressure injection, reactor building spray, core flood tanks and piping, decay heat removal, etc.). In addition, it was assumed that no piping under 2½ inches in diameter (small bore) had been dynamically analyzed by computer. Upon further review (including the licensee's submittals to the NRC dated July 30, September 5, October 31, 1979 and August 1 and October 10, 1980) by the NRC and SMUD, it was determined that the intent of the IEB 79-14 had not been met, in that there existed piping systems which were Seismic Category 1 systems as defined in the licensee's FSAR Appendix 5B which were not part of the previous scope of bulletin activities. Examples of such systems were: primary coolant system components, including vent and drain piping inside containment; main steam and feedwater piping up to the stop valves; atmospheric dump and main steam safety valves and associated piping from the main steam headers; and portions of the spent fuel cooling system. Also, it was determined that some small bore piping had been computer analyzed and had not been a part of the IEB 79-14 review. These systems include the RCS vent line piping and pressurizer vent line piping.

The portion of system piping which significantly contributed to the cause of the recent pipe leak was the nitrogen supply system. The original design of this system was a straight run of pipe without the use of a removal spool piece. The design had been dynamically analyzed by computer. A nonconformance report was initiated in 1974 to preclude the potential for contaminating the nitrogen system by designing a removal spool piece into the system to physically separate the nitrogen system and the RCS during normal operations. The addition of the removal spool piece required a change in the piping geometry in the vicinity of the spool piece addition. However, there were no engineering documents to show that the original analysis was revised and properly reviewed. This condition was postulated to have existed since 1974. Since the actual as-built condition (spool piece and piping geometry change) did not match the seismic analysis, if this line had been subjected to IEB 79-14 inspection as intended, there was reasonable assurance that the necessity of a replacement spool piece during operations would have been detected and corrected, possibly preventing the overstress condition which ultimately resulted in the piping failure.

b. During the previous IE Bulletin 79-14 activities, the licensee had described two instances in Reportable Occurrences 79-14 and 79-15 (transmitted to the NRC on November 9 and November 16, 1979 respectively) in which the as-built configuration did not match the seismic analysis. Analyses had indicated that in the as-built conditions, allowable design loads and stresses would be exceeded under seismic conditions. Additional pipe supports and the modification of existing supports were required. The associated Work Requests (39375, 40643, 43370, 43047) and support drawings were

reviewed and found to reflect the proper installation, modification and inspection of the affected supports.

c. The previous reanalyses of piping systems were reviewed to determine whether the Bechtel scope of responsibilities was properly implemented. From the SMUD submittal to the NRC (Region V) dated September 5, 1979, three stress calculations which required reanalysis were reviewed (stress calculation No. 8, 76, and 106). In addition, the computer input sheets for these three stress calculations were reviewed. The reanalyses were found, except in one instance, to properly reflect the as-built conditions determined in piping system walkdowns. In the one instance, it appeared that the stress isometric sketch for calculation No. 8/106 correctly located data point 185, but the computer input sheet was missing data point 185. Therefore, a short span (1'-10") of 2½ inch diameter piping was not correctly modelled.

In addition, in stress calculation No. 8/106, an 11 foot span of 3 inch diameter pipe was modelled without a midpoint data point (between data points 240 and 245). Since masses and stresses are calculated at the data points only, the omission of a data point could under-estimate piping stresses and support loads. In Bechtel's preliminary estimation, the midpoint data point could have been legitimately omitted from the analysis as the span's natural frequency was approximately 30-33 HZ. After the inspectors identified this concern, the stress calculation was redone using the proper coding of data point 185 and an intermediate data point in the 11 foot span of piping. The results of the re-analysis showed that pipe stresses remained low (significantly below stress allowables), and pipe support loads were within design allowable.

It was shown by reanalysis that the discrepancies identified in stress calculation 8/106 were of a minor nature in that stresses remained within design allowables. The inspectors concluded that, although some discrepancies were identified in the review of reanalyses done for IEB 79-14 walkdowns, no additional review was necessary based on the minimal effect on stresses and loads.

d. Two SMUD audits of the Bechtel activities for IEB 79-14 were reviewed. Audit No. 0-284 and 0-312, dated March 6 and July 22, 1980 respectively, were found to adequately verify the Bulletin responsibilities of Bechtel (i.e., preparation of stress problem package for all designated lines, evaluation of walkdown inspection results, reanalysis of lines if required, and proper disposition of discrepancies). Since it was not Bechtel's responsibility to designate the lines or systems to be covered under the Bulletin, these audits would not be expected to have identified the fact that there was an inadequate scope of review of systems as previously described.

e. The quality assurance organization conducted a surveillance of the engineering walkdown teams by independently verifying the as built condition of seven systems. The seven systems were selected by QA at random. The surveillance included inspection of the as built

drawings for pipe routing and correct pipe support location, reconciliation of the piping stress isometrics with support hanger drawings, and the evaluation of the engineering walkdown teams' findings by the QA organization. Seven lines were walked down as follows:

SYSTEM	CALCULATION NO.	LOCATION
Feedwater System	CAL #49b	Auxiliary Building
Reactor Coolant System	CAL #98	Containment Building
Seal Water Injection System	CAL #45	Containment Building
Letdown System	CAL #113	Auxiliary Building
Main Steam System	CAL #201	Turbine Generator Bldg
Borated Water System	CAL #75	Tank Farm/Auxiliary Bldg
Reactor Coolant Drain System	CAL #120	Containment Building

The quality assurance walkdown was accomplished with assistance from stress analysts from the Stone & Webster Corporation. The seven packages reviewed were originally walked down by six engineering walkdown teams; therefore, the sample reviewed by the quality assurance organization was comprised of twelve members of the engineering walkdown team.

The QA surveillance identified a discrepancy with the engineering walkdown teams' reviews. This discrepancy concerned a support on the 10" main steam line to the turbine generator reheater. The specific problem involved a support which was restraining the line in a direction which was not assumed in the original stress analysis. The discovery of this restraint in the improper direction was not identified by the engineering walkdown team because the hanger drawing which was used for construction and for the walkdowns contained several details of standard hangers used at the site. However, only one of the standard details which appear on the hanger drawing was to be installed. The other typical support configurations shown on the drawing were not to be used. Apparently, during construction also, the correct hanger to be used was not clear, and, therefore, an incorrect configuration was fabricated and installed.

This discovery by the quality assurance audit was made by checking the hanger drawing against the piping stress isometric. The piping stress isometric did show the directions in which the pipe was to be restrained as assumed by the stress analysis. The problem which surfaced appears to have been caused by the use of a standard support hanger drawing by the walkdown team which contained several standard support configurations but only one of the configurations was to be used. The support configuration to be used was labelled with a title and therefore should have been the only configuration used. Neither the seismic stress isometric nor the labelled hanger drawing detail was taken to the field by the field walkdown team.

As corrective action for this deficiency, identified by quality assurance, the engineering organization began a re-review of all

packages which were issued to the engineering walkdown teams. This review was to identify any similar support drawings which show standard details, are not labelled, and therefore, should not have been fabricated and installed. If one of these support drawings should be discovered, licensee personnel stated that a reinspection will be performed to verify that only the correct support was installed.

The inspector concluded that the QA surveillance appeared rigorous and it identified a significant finding which was being followed up.

f. The licensee generated a procedure for controlling the review and dispositioning of nonconformance reports generated by the engineering walkdown team. This procedure (#ERPT-M0002) described the steps which the stress analyst/pipe support designer took to review findings identified by the engineering walkdown team. Responsibilities for the individuals participating in the review were defined and established responsibilities of the supervising engineer, the initiating engineer who performed necessary calculations to resolve any nonconformances, and the checking engineer who would verify the assumptions and completeness of the initiating engineer's work.

The procedure also included the piping evaluation criteria, such as the analysis methodology, modeling techniques, and computer program to be used (the standard Bechtel pipe stress analysis code ME101 Version K 2). Also included were the Piping Overlap Method to be used (should it be necessary because of extremely large piping), and acceptance criteria such as allowable piping stress, nozzle loads, valve accelerations, and pipe support load criteria. Load combination tables for the analysis were also contained within the procedure.

The inspector concluded that the procedure provided the necessary guidance and information to the analyst to perform the evaluation in a controlled fashion.

The inspector interviewed three engineers involved in the NCR review process. The individuals interviewed were performing in process evaluations of discrepancies identified by the engineering walkdown team on the following three systems: Main Steam to the Main Feedwater Pump, the Reactor Coolant Drain System, and the Feedwater System. Topics discussed included the specific work the engineer was evaluating, such as redesign of a pipe support or reanalysis of a piping system to determine maximum postulated stresses within the piping system. Discussions were held with a pipe support design engineer, two stress analysts, and four team group leaders. During these discussions the in process work was discussed such as applicable piping codes, modeling techniques, applicable welding codes, nozzle allowable loads for large equipment such as pumps and heat exchangers, and geometry data input to the piping stress analysis computer code.

The inspector determined that all engineers were familiar with the applicable codes used and analytical techniques employed for this reanalysis. The stress analysis being performed for the main steamline was adequate and in agreement with the licensee's procedure and design criteria documents and industry practice for the portion reviewed.

The Rancho Seco plant was originally designed under the rules of ANSI-B31.7. During the review phase, the licensee elected to perform their review under the rules of ASME Section III. The licensee has elected to use a more current code in order to expedite a more efficient review process, in that procurement of pipe support parts under the rules of ASME were readily available. In addition, the engineers performing the review were more familiar with the rules of ASME (current day piping code) than the rules of ANSI-B31.7 (a piping code no longer used in the design of nuclear power plants). This decision by the licensee would also increase the conservatism of the analysis slightly, due to the increased requirements within the ASME Code on stress intensification factors used for various pipe fittings such as tees.

The licensee indicated that some of their reanalyses may require use of Code Case N411. The licensee has requested that the Office of Nuclear Reactor Regulation approve the use of this Code Case.

The inspector concluded that performing these analysis under the rules of ASME Section III was adequate for the IE Bulletin 79-14 analyses which were performed.

g. To substantiate high cycle fatigue failure of the "B" hot leg event, the licensee performed a hand calculation to assess the fatigue life of the "B" hot leg vent. This calculation was performed to substantiate the findings that the "B" hot leg vent mode of failure was high cycle fatigue due to a improperly supported small bore line.

The licensee elected to use a procedure for calculating the vibrational fatigue life of the piping system as described in ANSI/ASME OM3-1982, titled "Requirements for Preoperational and Initial Startup Vibration Testing of Nuclear Power Plant Piping Systems". Using this document as guidance, the licensee then established a procedure for determining the permissible number of cycles before failure would be obtained. This procedure required assumptions to be made on the predominate frequency of the piping system. The frequency is needed in order to determine the fatigue life of the piping system.

The predominant frequency was calculated by the licensee from the dynamic analysis performed on the piping system. Therefore, the value generated by the computer program was a natural frequency of the piping system at rest rather than the predominant frequency of system in operation. Currently, no data exist for the Rancho Seco facility which would allow the determination of the predominant frequency of the system with all reactor coolant pumps running.

Therefore, the licensee has utilized a best estimate approach of using the natural frequency of the piping system.

The inspector concluded that although this method for determining fatigue life of the "B" Hot Leg Vent was not the most rigorous method available, it was probably the only method available with the limited amount of data currently available.

The licensee's analysis showed that acceleration levels for the "B" hot leg vent greater than 0.633g could produce stresses at the nozzle connection which were beyond the code stress limit for steady state vibration. The calculated acceleration would produce a deflection of the piping system of approximately 0.59 inches. The estimated deflection of the piping system in the as found condition after the leak had been discovered was approximately 0.9 inches. Therefore, the licensee concluded that the maximum deflection the piping system could tolerate from an assumed acceleration was exceeded with the as found piping configuration.

The inspector concluded that the piping system, as found, might not have tolerated the current configuration without ultimate failure of the piping system. However, this calculation method was a very simplified approach, and would only yield preliminary results. This method supported previous conclusions by the licensee that failure of the hot leg vent was caused by high cycle fatigue.

6. Conclusion

The inspection clearly indicated that a quality assurance system failure had occurred in the "B" high point vent's design, construction, and inspection. Necessary design information was not properly included in the design package even though this information was detailed on the pipe stress isometric as well as the October 7, 1981 letter (mentioned above) that were transmitted to the licensee. Lack of control over the construction work that installed the modification was evidenced by the omission of several pipe supports. The quality control inspection performed was not detailed enough to identify the conflict between the as-built configuration and the design. Finally, the quality assurance organization's program of review and surveillance did not identify any of the deficiencies that plagued the "B" high point vent modification.

The nitrogen supply and vent header systems connecting to a one inch reactor coolant system vent located on the high point of the "B" hot leg were not identified as systems to be covered under the licensee's QA program required by 10 CFR 50, Appendix "B". These systems, therefore, were not designed, constructed and inspected pursuant to QA Class I requirements.

On June 23, 1985, a crack was identified in the above mentioned reactor coolant vent line that resulted in a 17 gpm primary coolant leak that was non-isolable. The cause of this condition was determined to be fatigue from high cycle vibration, induced by the nitrogen and vent header systems because of design and construction errors in 1974, 1981, and 1983

that resulted in missing pipe supports, improper pipe supports and a missing pipe section as described below:

- missing cross-brace between lines 20574-1"-HE and 20555-½"-CA
- missing rigid flanged spool piece that should have been installed in the nitrogen supply line during operation of the plant
- missing east-west stops on support number 1S20567-3
- missing east-west stops on support number 1S20567-2
- no replacement of vertical spring support with required stops on support number 1S20567-2

The above deficiencies in the systems were identified in a stress analysis performed by the licensee's contractor in 1981 in preparation for addition of the high point vent modifications which were made to the system in 1983. The required piping supports were not installed, even though the need for these piping supports was transmitted to the licensee by letter from their design contractor on October 7, 1981.

In addition, the design drawings for construction of the reactor coolant system 3/4 inch high point vent lines on both the "A" and "B" sides specified seismic frames located between the two sets of solenoid valves, but these supports were found missing in June 1985. Numerous additional examples of similar deficiencies in safety-related systems were identified and documented in the licensee report to NRC's Region V office dated August 6, 1985.

Similar concerns had been identified by previous inspections. The licensee demonstrated an inadequate control of the design and installation of auxiliary feedwater flow transmitters in Inspection Report 84-15. Inadequate construction and ineffective quality control inspections associated with the Nuclear Service Electric Building were discussed in Inspection Report 85-01.

The inspectors concluded that the previous and present findings had showed the same pattern of performance in the engineering, construction, quality control and quality assurance areas.

Enforcement action related to the issues discussed in this report will be the subject of separate correspondence.

7. Exit Meeting

The inspector met with licensee management personnel denoted in paragraph 1 at the conclusion of the inspection on August 9, 1985. The scope, observations, and findings of the inspection were discussed.