EXHIBIT A

JOHN MERTENS

Mr. J. B. Martin
Regional Administrator
U. S. Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region V
1450 Maria Lane, Suite 210
Walnut Creek, California 94596-5368

July 20, 1983

Dear Mr. Martin:

The purpose of this letter is to bring to your attention technical inadequacies in safety related piping systems and other recent incidents at the San Onofre Nuclear Power Station. They are itemized below. For a detailed discussion please see the remainder of this letter and the attached copy of an engineering evaluation I recently completed.

1. Evidence indicates that Unit 2 operated for 21 months with inoperable snubbers on the safety related main feed water line FW 189 (from March 1981 through February 1982 with six, and from March 1981 through December 1982 with five internally damaged snubbers).

2. Inspections of these snubbers carried out over those periods failed to detect their inoperability.

3. An attempt was made to obstruct the engineering evaluation I conducted.

4. The method presently used for inspecting snubbers is inadequate and misleading.

3109 S. El Camino Real · San Clemente, CA 92672 · (714) 498-6311

5. No effective QA program exists for monitoring snubber performance as per Technical Specification page B3/4 7-6, Snubber Basis.

6. A recommandation to install minimum instrumentation for monitoring operability of snubbers in safety related piping was dismissed.

7. The engineering evaluation performed by Bechtel Power Corporation to determine the cause of failure of these snubbers is inadequate and misleading.

8. Snubbers on the main feed water lines inside the containment were not designed for dynamic loads, they are undersized and may not be able to ensure structural integraty of the main feed water lines.

9. An attempt was made to intimidate me (transferred or fired) in response to my questioning BPC's performance at Songs 2 & 3.

10. The original and all copies of NCR S01-P-1308, Rev. 1, were destroyed.

As to my qualifications for reporting such matters, I am a station engineer here at Songs., I am 58 years old and have extensive experience in design and development work, and failure analysis.

One of my recent assignments was analysing why five snubbers on the main feed water line, FW 189, Unit 2, inside the containment at elevation 63', hanger locations H010, H013, H017 had become inoperable. Suspected inoperability was first reported on December 8, 1982.

I had the five snubbers shipped to the manufacturer, Pacific Scientific, for disassembly and inspection. Their internal parts were found severely damaged. The sixth snubber, which was the twin of one of the damaged ones, had been stroked at the time the others were removed and found operational. This, however, seemed irreconcilable in the light of the severely damaged five snubbers. There were two explanations: one, the sixth snubber was not in place when the others were damaged, and second, the sixth snubber had somehow survived the destructive event.

It became necessary, in my judgement (T/S 3/4.7.6 (g) 1st pgr.), to remove the sixth snubber for two reasons: first, to ascertain operability through functional testing and internal inspection by the manufacturer, thereby ensuring Unit 2 was not operating with a damaged snubber on the main feed water line, and two, if the sixth snubber was found operational, it would be a clue as to when the damaging event had occured.

My immediate supervisor agreed with this reasoning. I submitted then a work order to Startup Maintenance Support requesting removal of the sixth snubber. It was cancelled. And so were two further work orders. As to the fourth one, some one called Startup Maintenance Support in my name and instructed them to cancel it. I then submitted a new NCR to accomplish removal of that snubber. Yet, even that was recommended for cancellation by Project personnel. Only on my insistance was that sixth snubber removed. It was taken to the manufacturer for functional testing and internal inspection. It was found to be operational. This was convincing evidence that the sixth snubber had been installed after the damaging event.

Researching records suggested that the original sixth snubber (SN 2609) was replaced by SN 4322 on February 1, 1982. I say suggested because NCR S023-F-463 required rejection of the damaged snubber, but the respective work order (2113) reports only that a new tag was attached. No record of SN 4322 appears until 12-23-82 on work order 19773.

On 4-26-83 snubber SN 2609, the original sixth snubber, was retrieved from the Bechtel Warehouse. An examination showed that it was as badly damaged as the other five. This confirmed that the damaging transient had occured before February of 1982, and that the causing event was the Waterhammer Test 2HA-201-01 of January 30, 1981.

A final test conducted at the Pacific Scientific Laboratory revealed that the PSA-10 snubber fails under shock loading at 37,000 lbs. But more important was the revelation that the destruct tested snubbers showed no visible evidence of failure, and further, they could be stroked without indication that the internals were damaged.

The disturbing element in these conslusions is the implication that: a) Unit 2 operated for 21 months with damaged snubbers while going through modes 4 - 1, b) visual inspections of snubbers are unreliable, c) QA and maintenance work as related to snubbers is inadequate, d) no effective means exist to detect damaged snubbers.

The need for a better method of surveillance for snubbers became obvious. Therefore, I submitted a written recommendation to install minimum instrumentation on the main feed water lines for monitoring snubber conditons, transients, and for alerting operating personnel when a transient had occured. With such instrumentation operability of snubbers and respective systems can be checked quickly. As of today no such instrumentation has been considered as neccessary.

Another matter, which is of personnel concern, arose when I questioned Bechtel's performance here at Songs. I discussed this with other engineers who advised me that previous questioning of Bechtel's performance had been countered by SCE management with a message that had quenched further inquiries. I decided,therefore, to pursue the matter as a stockholder. After an exchange of fruitless correspondence with the General Office the technical station manager called me to his office and told me on instructions from top management that if I continued my inquiries into these matters I would either be transferred or fired. A note that I had been counseled in certain matters was put into my performance report (se attached copy).

Because of these incidents I request that you take appropriate action. I also recommend for your consideration the exclusion of BPC from performing any engineering failure evaluations for Units 1, 2 and 3 at San Onofre Nuclear Power Generating Station because of conflict of interest. All investigative avenues that were pursued for the attached engineering evaluation were open to BPC. They chose to ignore them.

Please inform me as to what action you will take.

Sincerely

John Mertens

John Mertens

Enclosures

EXHIBIT B

Southern California Edison Company

P. O. BOX 800

2244 WALNUT GROVE AVENUE

ROSEMEAD, CALIFORNIA 91770

DAVID J. FOGARTY
EXECUTIVE VICE PRESIDENT

TELEPHONE
213-572-2798

July 27, 1983

Mr. John Mertens
3109 South El Camino Real
San Clemente, California 92672

Re: San Onofre Nuclear Generating Station,
Units 1, 2 and 3

Dear Mr. Mertens:

Your letter to Mr. Gould concerning alleged technical inadequacies in safety-related piping systems, dated July 20, 1983, was received on July 22 and referred to me for evaluation and response. You requested a response by July 27, 1983, concerning whether or not a letter from you to the Nuclear Regulatory Commission (NRC) contained false statements.

The time available to respond to you is very short, and I request that you give us until August 5, 1983, to respond so that our response can be complete and the details can be checked. In addition, I urge you to follow the procedure described in Mr. Gould's letter dated July 11, 1983 (copy attached), for pursuing concerns about nuclear safety. (This letter updated Mr. Gould's letter of September 13, 1982, which described this procedure to all personnel.) I have directed the Onsite Review Committee to proceed with review of your nuclear safety concerns in any event. Please let me know if we may have until August 5 to respond and if you will agree to pursue your nuclear safety concerns in accordance with Mr. Gould's letter.

Nevertheless, a response to your request has been developed in the time available. Attached hereto are our comments on the content of your proposed letter to the NRC. A best effort has been made to investigate the facts and circumstances involved; however, the response must be considered preliminary.

In summary, as described further in the attachments, the technical matters related to your nuclear safety concerns are adequately identified and documented within our design, testing and quality assurance programs at San Onofre.

Sincerely,

D. J. Fogarty

D. J. Fogarty
Executive Vice President

DJF:CRK:dkg
Attachments

Southern California Edison Company

P. O. BOX 800

2244 WALNUT GROVE AVENUE

ROSEMEAD, CALIFORNIA 91770

WILLIAM R. GOULD
CHAIRMAN OF THE BOARD

TELEPHONE
213-572-2288

July 11, 1983

TO ALL PERSONNEL
ADMINISTRATION
ADVANCED ENGINEERING
ENGINEERING & CONSTRUCTION
FUEL SUPPLY
NUCLEAR ENGINEERING AND OPERATIONS
POWER SUPPLY
SYSTEM DEVELOPMENT

SUBJECT: Review Process for Nuclear Safety Concerns

Purpose

To remind personnel of the existence of a review process which is available to address nuclear safety questions and concerns.

Discussion

Since 1967 when the Company received an operating license for San Onofre Unit 1, safety review organizations have been in effect whose functions include review and initiation of action to solve problems related to nuclear safety at San Onofre. These organizations now include the "On-Site Review Committee" located at San Onofre and, at the General Office, the Nuclear Audit and Review Committee" for Unit 1 and the "Nuclear Safety Group" for Units 2&3. These organizations are available to consider questions or concerns related to nuclear safety from employees who become aware of existing or potentially serious problems.

In keeping with long-standing practice, should you become aware of a nuclear safety question related to a Company facility, the matter should be brought to the attention of your supervisor. He or she will, in consultation with the proper personnel, resolve the question and advise you of the resolution if you so request. If you do not feel the question or concern was satisfactorily resolved, you should notify the head of the appropriate safety review organization. For operations and maintenance personnel working at San Onofre this is the Chairman of the On-Site Review Committee and for other personnel this is the Chairman of the Nuclear Audit and Review Committee for Unit 1 matters, or the Manager, Nuclear Engineering and Safety for Units 2&3 matters.

In order to insure your concern is properly understood and that you are informed of the outcome, this notification should consist of correspondence from you describing your concern.

Following review of your concern you will be notified of the outcome. You should also be aware that the records of these actions are subject to audit by the Nuclear Regulatory Commission, and the disposition of your concern will be made a part of those records.

As you know, we are relying on nuclear generating stations for a significant share of our future generating capacity and are dedicated to maintaining our exemplary nuclear safety record. To accomplish this, it is important that potential nuclear safety questions be identified and promptly resolved. Each employee involved in the Company's nuclear program should consider nuclear safety and compliance with NRC regulations as the first priority in executing their duties.

This procedure is in addition to and not in derogation of any rights or obligations provided under the regulations of the Nuclear Regulatory Commission.

William R. Gould

COMMENTS ON PROPOSED LETTER DATED
JULY 20, 1983, FROM JOHN MERTENS TO
J. B. MARTIN, USNRC

The proposed letter refers to ". . . technical inadequacies in safety related piping systems and other recent incidents at the . . ." San Onofre Nuclear Generating Station (SONGS). Comments, based on a preliminary review of information immediately available, are provided below for the items listed in the proposed letter to Mr. Martin.

Item 1: Evidence indicates that Unit 2 operated with several mechanical snubbers not detected as inoperable in main feed water line FW189.

Comment: As developed in Mr. Mertens' report to Mr. Katz dated May 23, 1983, this certainly did occur for some period, and it is likely that the mechanical shock arresters (snubbers) discussed were initially damaged in the March 21, 1981, startup test indicated. It should be noted that operability of main feed water line FW 189 was first required for Unit 2 when it entered Mode 3 on May 18, 1982.

The startup testing program for Unit 2 included monitoring this line for thermal expansion, and it noted an anomaly which resulted in snubber inspection at the 50% power plateau in December, 1982. This inspection did identify snubber damage and led to the replacement of five snubbers at that time. Replacement of the first two of these was reported to the NRC in LER 82-165. This LER is being revised to include the three additional snubbers replaced during that same time.

Item 2: Inspections of mechanical snubbers in main feed water line FW189 between when the damage probably occurred in March, 1981, and when it was discovered in December, 1982, did not detect the inoperability.

Comment: Pre-fuel loading snubber inspections in January, 1982, did identify two damaged snubbers on this line, which were replaced. Damage to other snubbers was not identified until December, 1982. During the period from January to December, 1982, a number of plant transients occurred which could have caused, or increased, damage to these snubbers.

Item 3: An attempt was made to obstruct the engineering evaluation assigned to Mr. Mertens.

Comment: This appears to relate to difficulty experienced in March, 1983, in removing the snubber installed in January, 1982. As indicated, Mr. Mertens' supervisor supported this effort, which was successful when the appropriate documentation (an NCR) was prepared. This snubber was found to be undamaged and operable.

Item 4: Mechanical snubber inspection methods are inadequate and misleading.

Comment: Snubber inspection methods include periodic 100% visual inspection and stroke testing of selected snubbers. In addition, a sample of the snubbers is periodically removed for force testing by machine. The sample is enlarged when problems are noted. Each of these inspection methods is capable of identifying deficiencies, as demonstrated by extensive experience.

Item 5: No effective quality assurance program exists for monitoring snubber performance in accordance with the basis for snubber operability described on page B 3/4 7-6 of the Technical Specifications.

Comment: Page B 3/4 7-6 of the Technical Specifications discusses the requirements for monitoring the service life of snubbers subject to environmental degradation (i.e., hydraulic snubbers).

Presently, an effective program, implemented by procedures, is in place to monitor both hydraulic and mechanical snubbers. This program consists of both a visual and functional test to ensure proper snubber performance. A computer-based system which records the maintenance history of all snubbers is utilized to input information into the snubber surveillance program. Any failures that are identified during these inspections are subjected to a detailed engineering analysis and, if determined necessary, an increase in the frequency of inspection or the number of snubbers inspected is implemented.

Item 6: A recommendation from Mr. Mertens to install minimum instrumentation for monitoring the operability of snubbers in safety-related piping was dismissed.

Comment: Mr. Mertens participated in a meeting between representatives of NUS Corporation, the Electric Power

Research Institute (EPRI) and SONGS in early June, 1983, which included discussion of this recommendation. Pursuant to that meeting, SONGS expects a proposal from NUS in early August, 1983, which will include the possibility of use of monitoring instrumentation. Also, on July 12, 1983, Mr. Mertens' report of May 23, 1983, which includes this recommendation, was forwarded for evaluation by Bechtel.

Item 7: The engineering evaluation performed by Bechtel to determine the cause of failure of snubbers in the main feed water line FW189 was inadequate and misleading.

Comment: The evaluation performed initially was primarily concerned with verifying that the line was not overstressed as a result of the event which damaged the snubbers. This was established in an adequate and straightforward evaluation. The initial Bechtel evaluation did not definitely establish the cause of the event, and Bechtel was requested to perform further evaluations in letters from the SCE Project Engineer dated February 3, 1983 and April 7, 1983. Funding for this further evaluation was approved by SCE on May 18, 1983, and the work is ongoing. As the work is not complete, it is incorrect to characterize it as inadequate and misleading.

Item 8: Snubbers on the main feed water lines inside the containment were not designed for dynamic loads, they are undersized and may not be able to ensure structural integrity of the main feed water lines.

Comment: Main feed water lines inside the containment, including their supports, are designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class 2. The design has been verified as correct. As indicated in the design basis (FSAR paragraph 10.4.7.1.d), it does not include dynamic loading resulting from severe water hammer transients, as these are to be avoided by operational and design measures. The startup test on March 21, 1981, which is hypothesized to have caused the damage to the five snubbers which was revealed by the inspections of December, 1982, and to the two snubbers which were replaced in January, 1982, was substantially more severe than is included within the design basis. Further, following steam generator feed ring modifications in mid-1981, a revised startup test was successfully conducted in June 1982 which confirmed the adequacy of the main feed system design basis. In summary, the snubber damage probably occurred during startup testing, which included a severe water hammer transient. Should such a transient beyond the design basis reoccur during plant operation, careful inspection and testing of affected snubbers would be performed including consideration of this startup testing experience. Even so, analysis shows that feed water line FW189 was not overstressed by the event which damaged the snubbers.

Item 9: An attempt was made to intimidate Mr. Mertens (transfer or dismissal) in response to questoning Bechtel's performance at SONGS 2 and 3.

Comment: No attempt was made to intimidate Mr. Mertens. In an exchange of correspondence between Mr. Mertens, SCE and others commencing in November, 1982, he requested:

- Total expenditure data concerning SONGS 2 and 3

- Payments to Bechtel for engineering work on SONGS 2 and 3

- A copy of the contract between SCE and Bechtel for SONGS 2 and 3

- The same information as above for SONGS 1

Mr. Mertens pursued these documents and data as a stockholder with the Secretary's office, rather than as an employee. In his letters, he included the comment that the reason for his request ". . . is to determine whether or not the [contract] contains provisions for redress in the event of unsatisfactory performance, and to study the terms of such provisions." At no time prior to receipt of his letter to Mr. Gould dated July 20, 1983, were questions of nuclear safety raised in this correspondence.

Mr. Mertens was counseled in February, 1983, to stop his repeated attempts to obtain the information above, as SCE has declined to provide it. This counseling did not adversely affect his overall performance rating in his March, 1983, performance appraisal, and he was not threatened with transfer or dismissal

Item 10: The original and all copies of NCR S01-P-1308, Rev. 1, were destroyed.

Comment: Revision 0 of this NCR for SONGS 1 was opened on December 8, 1982. The condition described relates to corrosion of a heat exchanger foundation and is also described in an NRC inspection report transmitted to SCE by NRC letter dated January 7, 1983. The work required to repair the foundation is ongoing and is about 80% complete. Revision 1 of the NCR was drafted to readdress the original evaluation of heat exchanger operability which had been made for Revision 0. Some difficulty was experienced in deciding how to treat operability during the repair. It was concluded prior to validating the NCR revision that readdressing operability was not necessary, and the repair could be made with the heat exchanger in service, so the revision was not completed. Revision 0 of the NCR remains in effect, and it governs the work which is now nearing completion.

In addition to the ten items identified and commented on above, the letter discusses Mr. Mertens' assignment to investigate the cause of the failure of snubbers in main feed water line FW189. This assignment was made December 23, 1982, as a result of a station incident report written to cover the matter. The assignment resulted in his report to Mr. Katz dated May 23, 1983, which was forwarded for evaluation by Bechtel via a startup problem report dated July 12, 1983. At the same time, as the result of a series of exchanges between SCE project engineering and Bechtel, further evaluations of the cause of the event resulting in the snubber damage was approved on May 18, 1983, and is ongoing.

EXHIBIT C

Southern California Edison Company

SAN ONOFRE NUCLEAR GENERATING STATION
P.O. BOX 128
SAN CLEMENTE, CALIFORNIA 92672

H. B. RAY
STATION MANAGER

TELEPHONE
(714) 492-7700

August 5, 1983

Mr. John Mertens
3109 South El Camino Real
San Clemente, California 92672

Re: San Onofre Nuclear Generating Station,
Units 1, 2 and 3

Dear Mr. Mertens:

Mr. Fogarty's letter to you dated July 27, 1983, provided our initial response to your letter to Mr. Gould dated July 20, 1983. Attached hereto for your information is our more complete response, as described in Mr. Fogarty's letter.

Also, the Onsite Review Committee (OSRC) at San Onofre has met to review those of your concerns which relate to nuclear safety. We have had the benefit of your participation in clarifying those concerns. OSRC has provided input to the attached response and will review the conclusions of actions which are not yet completed.

In summary, we conclude that adequate means exist to ensure that your concerns which relate to nuclear safety are identified, documented, evaluated and appropriately resolved. The OSRC will continue its oversight, and you are encouraged to bring additional information to the committee and to participate in its further review.

Sincerely,

H B Ray

Attachment

I. SUMMARY

In a letter to Mr. W. R. Gould, dated July 20, 1983 (Attachment 1), an employee of Southern California Edison Company described certain concerns regarding safety-related piping systems at San Onofre Nuclear Generating Station. Southern California Edison provided a preliminary response to those concerns in a letter, dated July 27, 1983 (Attachment 2), from Mr. D. J. Fogarty to Mr. J. Mertens, the involved employee. That letter indicated a complete response would be provided to Mr. Mertens by August 5, 1983. This report provides that complete response.

In order to provide the background necessary to understand the concerns raised by Mr. Mertens, the operational history of the main feedwater piping in question is discussed in Section II. Following the background information, each of his concerns is repeated verbatim in Section III. SCE's responses are provided after each item.

On July 28, 1983 and August 4, 1983, special Onsite Review Committee meetings, attended by the NRC, were held so that Mr. Mertens could present his concerns as they relate to nuclear safety. Section IV of this report repeats the concerns as they were described in these meetings (References 1 and 2), and provides the results of the committee's review. Section V summarizes those activities still in progress at the time of this report.

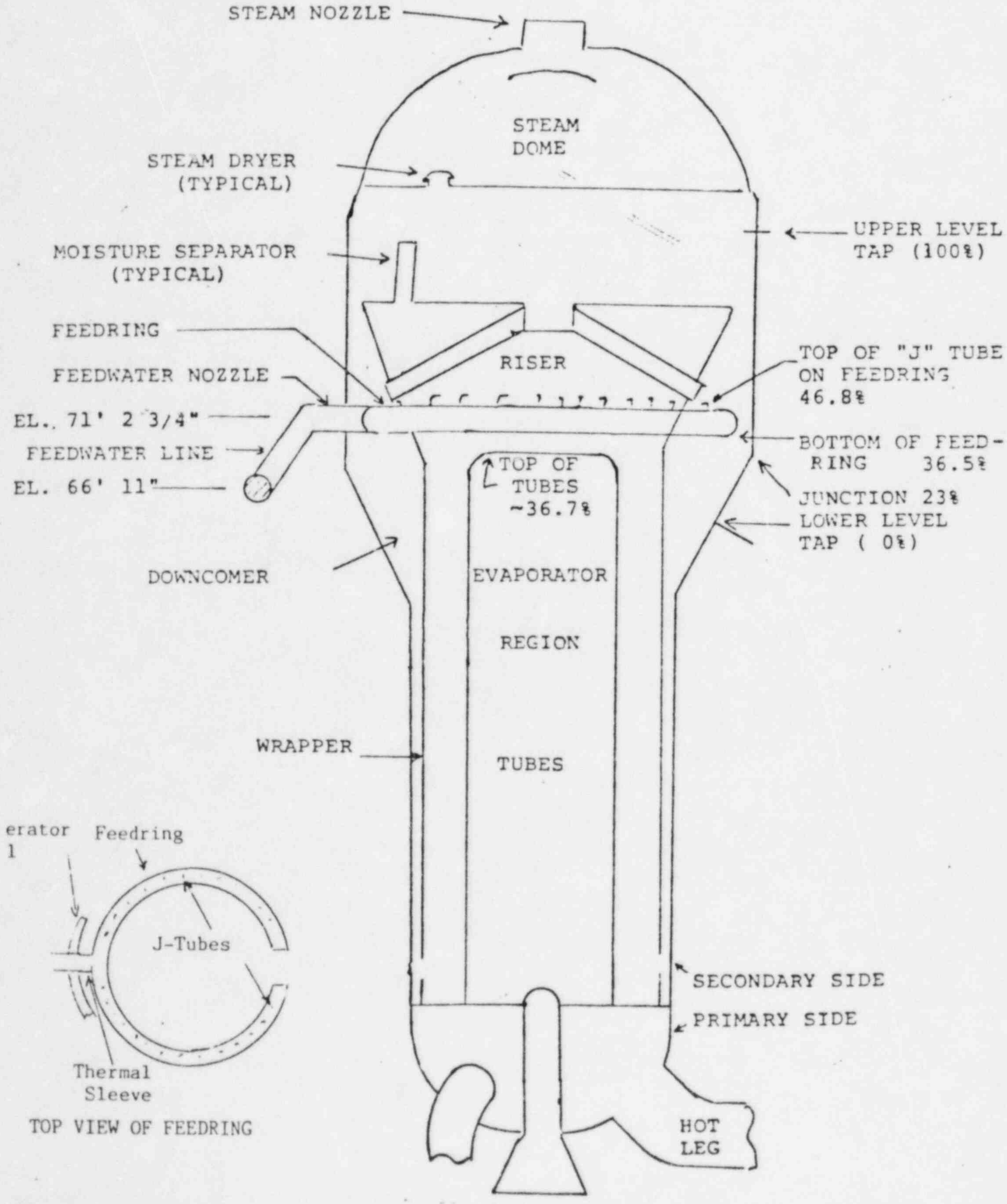
II. OPERATIONAL HISTORY OF MAIN FEEDWATER PIPING

The operational history of the main feedwater piping begins with the NRC's review of the piping design criteria. As stated in the Final Safety Analysis, paragraph 10.4.7.1.D, the system is not designed for abnormal hydraulic loads since design and operating considerations preclude their occurrence. The NRC requested that a special test be added to the startup program to verify the FSA design criteria in that unacceptable waterhammer would not occur as a result of uncovering and draining the feedring (refer to Figure 1 for a sketch of the steam generator internals). In March 1981, when this test was performed on steam generator E088 and feedline FW189, a thud was heard and attributed to a check valve slamming open. No damage to piping or supports was noted during a subsequent visual inspection of the piping. Later, in July 1981, during a scheduled generator internals inspection, the feedring was found to be damaged. The cause of failure was attributed to the dynamic differential pressure forces applied to the feedring due to rapid injection of auxiliary feedwater into the drained feedring. The design was modified to increase the time required to drain the feedring, to reduce the magnitude of the differential pressure dynamic loading, and improve the capability to withstand compressive loading. A supplemental feedring integrity test with less severe conditions was scheduled for later in the startup program to verify these modifications. This test was conducted in June 1982. No indications of waterhammer, such as noise or vibration, were observed during the test. A subsequent visual inspection of the feedring, the feedwater piping, and piping supports showed no damage.

FIGURE 1

LOCATION OF FEEDRING IN

STEAM GENERATOR -SONGS 2& 3



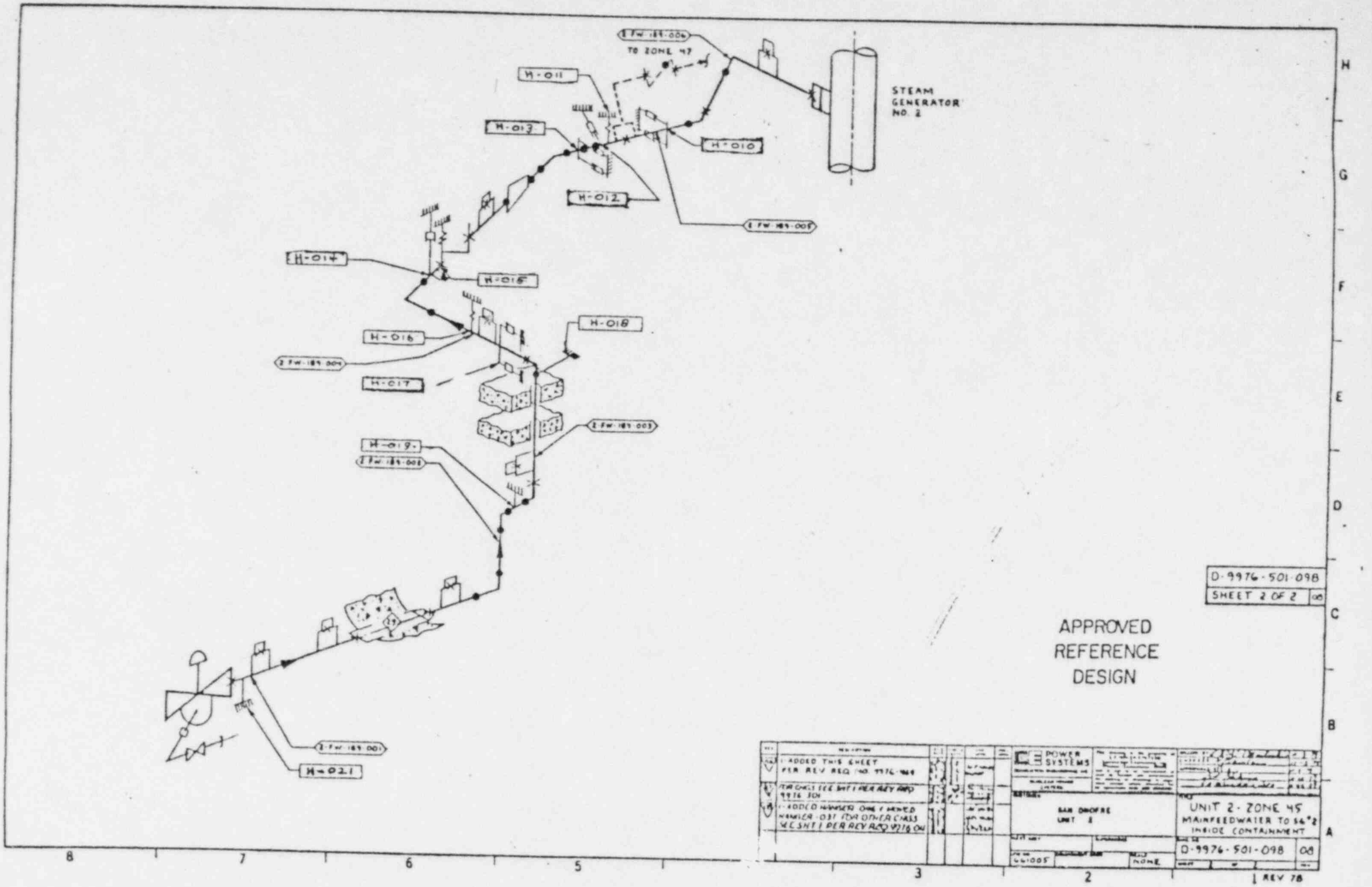
In January 1982, a pre-fuel load visual inspection was conducted of all safety-related snubbers. Two mechanical snubbers on FW189 failed this inspection when they would not rotate in place. The failure was attributed to an installation error and the affected snubbers were replaced. (Figure 2 is a diagram of FW189 inside containment, and Table 1 lists the history of the mechanical snubbers for FW189.)

In May 1982, during post-core hot functional testing, excessive vibration was noted in the auxiliary feedwater piping inside and outside containment on three separate occasions. A special test program was conducted in June 1982 to confirm the cause of the vibrations. Vibration magnitudes of the auxiliary feedwater piping inside containment were monitored and found acceptable. The program (see Attachment 3) confirmed the cause of the vibrations was not steam generator waterhammer, but was caused by hydraulic instabilities induced by backflow through a 2-inch Y-type globe valve located outside containment.

On November 9, 1982, the reactor tripped from 20% power due to low steam generator level caused by loss of feedwater control (Reference 3). Steam generator E089 level dropped below the feedring for approximately 10 minutes. Level was restored utilizing auxiliary feedwater pumps. These transient conditions were bounded by the conditions of the supplemental feedring integrity test described above, thus, further investigation into this event is not required.

On November 13, 1982 and November 26, 1982, the reactor tripped from high steam generator level (References 4 and 5). Steam generator feedline waterhammer was not a concern in either of these events since the feedring was not uncovered.

FIGURE 2



STEAM GENERATOR NO. 2
TO ZONE 47
H-011
H-013
H-010
H-012
H-014
H-015
H-016
H-017
H-018
H-019
H-021
D-9976-501-098
SHEET 2 OF 2
APPROVED REFERENCE DESIGN
POWER SYSTEMS
UNIT 2 - ZONE 45
MAINFEEDWATER TO S.G. 2
INSIDE CONTAINMENT
SAN ONOFRE
UNIT 2
NONE
D-9976-501-098
08
ADDED THIS SHEET PER REV REQ NO. 9976-
REV 78

TABLE 1

MAIN FEEDWATER LINE S2FW189 - MECHANICAL SNUBBER HISTORY

Location	Initial Snubber Serial Number	January 1982 Inspection	December 1982 Inspection	April 1983 Inspection
S2-FW189-H010 (Dual Snubbers)	2609	2609* replaced with 4322	4322	4322 replaced with 15371
	2603	2603	2603* replaced with 11077	11077
S2-FW189-012	390	390* replaced with 1145	1145	1145
S2-FW189-H013 (Dual Snubbers)	2113	2113	2113* replaced with 3472	3472
	2107	2107	2107* replaced with 268	268
S2-FW189-H014	3098	3098	3098	3098
S2-FW189-H017	4357	4357	4357* replaced with 11086	11086
	4352	4352	4352* replaced with 11085	11085

Note(*): Snubbers found to be damaged

On December 9, 1982, while performing the feedwater piping thermal expansion test at the 50% power plateau, a mechanical snubber at location S2-189-H013 (see Figure 2) was found damaged. The initial engineering investigation (Attachment 4) concluded that the remaining mechanical snubbers, five additional for a total of six in question, at locations S2-189-H013, S2-189-H010, and S2-189-H017 were probably also damaged. Eventually, the original damaged snubber and four others were replaced. The sixth, on support S2-189-H010, was manually stroked with satisfactory results and was not replaced.

Table 1 summarizes the status of each snubber at these supports. At the time of the event, a preliminary investigation into the failures postulated the cause to have been the November 9, 1982 transient discussed above.

As part of the followup review of this occurrence (Reference 6), Mr. Mertens was requested to evaluate the cooldown in light of the snubber damage report, confirm the postulated cause, and make recommendations for plant or procedural modifications to preclude recurrence. During this investigation, Mr. Mertens had the five failed snubbers, found in December 1982, shipped to the manufacturer for failure analysis. The manufacturer reported the cause of failure as gross overload (report included in Attachment 1). As a result of this report, Mr. Mertens recommended removal and inspection of the single snubber not replaced at location S2-189-H010. That snubber was replaced, sent to the manufacturer and passed a functional test and internals inspection.

Mr. Mertens reported the results of his investigation, the test results discussed above, and his conclusions in a memorandum to the Station Technical Manager, dated May 23, 1983 (included in Attachment 1). This memorandum was forwarded to Bechtel Power Corporation for further evaluation on July 12, 1983 (Reference 7). The evaluation of his memorandum is contained in Section III of this report.

III. RESPONSE TO CONCERNS STATED IN PROPOSED LETTER DATED JULY 20, 1983, FROM JOHN MERTENS TO J.-B. MARTIN, USNRC

The proposed letter refers to ". . . technical inadequacies in safety related piping systems and other recent incidents at the . . ." San Onofre Nuclear Generating Station (SONGS). The responses are provided below for the items listed in the proposed letter to Mr. Martin.

Item 1: "Evidence indicates that Unit 2 operated for 21 months with inoperable snubbers on the safety-related main feedwater line FW 189 (from March 1981 through February 1982 with six, and from March 1981 through December 1982 with five internally damaged snubbers)."

Reply: The initial evaluation of the inoperable snubbers found in December 1982 postulated the cause to be a transient occurring about one month earlier.

A more extensive evaluation of this event has been undertaken. Based on a review of the following data and as stated in our LER 82-165, Revision 2, it can now be concluded that initial damage to the snubbers on feedwater pipe FW189 occurred prior to the November 1982 cooldown events and probably as a result of the steam generator feedring integrity test conducted in March 1981:

Item 1: (Continued)

A. Dates of snubber inspections and replacement, and dates of linear main feedwater piping hydraulic transients discussed in Section II of this report.

B. Piping thermal expansion measurements, recorded per piping Thermal Expansion Test 2PA-102-01 match analytical results more closely than those taken at 20% power prior to snubber replacement in December 1982.

C. The other main feedwater pipe (FW-190) connected to the steam generator that did not undergo the feedring integrity test has exhibited acceptable thermal movements throughout the test program. Both main feedwater lines are moving as predicted by original design calculation since replacement of the damaged snubbers on line FW-189.

Additionally, since the calculation stresses as result of the event were found to be acceptable as discussed later in this report, no unacceptable loads were imposed on the piping system.

Item 1: (Continued)

Discovery of an inoperable snubber on main feedwater piping inside containment is reportable to the NRC. Therefore, the two primary methodologies for determination of reportability and operability, the Non-Conformance Report (NCR) and the Station Incident Report (SIR), were reviewed for proper application to feedwater pipe support problems. For the December 1982 snubber failure, three NCRs were issued and correctly classified as reportable and not operable. The engineering evaluation required by Technical Specification 4.7.6.g was specified in these NCRs and performed by Bechtel Power Corporation. A Station Incident Report was also prepared to document the snubber failures. This report also correctly determined the failure to be reportable. Therefore, it is concluded the reportability and operability assessments were made correctly.

The failure was reported to the NRC via LER 82-165 for two of the five snubbers. However, as a result of an administrative oversight, this LER had not been revised to reflect all of the failures. Revision 2 to LER 82-165, reflecting the additional damaged snubbers and conclusions with respect to the cause of damage has been submitted. Since SCE's April 1983 evaluation of the reactor trip breaker failures at Units 2 and 3 (Reference 9) resulted in improvements in the identification and implementation of the reportability requirements of the Technical Specifications, a repeat of this type of administrative oversight is not likely to recur and no further corrective measures are planned.

Item 2: "Inspection of these snubbers carried out over those periods [between when the damage probably occurred in March 1981, and when it was discovered in December 1982] failed to detect their inoperability." [The discussion in brackets has been added for clarity.]

Reply: Between March 1981 and December 1982, a number of snubber inspections of Feedwater Line FW-189 were conducted and are listed in Table 2. Since it is most likely that the feedwater snubbers were damaged in March 1981, Table 2 indicates that the visual and in-place rotation checks identified only 2 of the seven failed snubbers out of a total of eight snubbers on line FW189.

However, the thermal expansion test program and manual stroking did identify the rest of the failed snubbers. One snubber in support S2-FW-189-H010 was originally

Table 2

Inspections of Snubbers on Main Feed Line FW-189

Date	Reason for Inspection	Type	Findings
March - April 1981	Following Special S/U Test of S/G Feedwater integrity	Visual	No indications of damage to piping or supports
July 1981	Walkdown following discovery that March 1981 Feedring integrity test had collapsed the feedring	Visual	No indications of damage to piping or supports
December 1981 - January 1982	QA/QC Walkdown to verify snubber integrity prior to initial fuel load	Visual, Hands On (1)	o NCR F-462, snubber at S2-FW-189-H-012 could not be rotated, snubber #390 was replaced with #1145 o NCR F-463, snubber at S2-FW-189-H-010 could not be rotated, snubber #2609 was replaced with #4322 o No other indications of snubber damage were noted
November 1982	Walkdown following excessive feedwater addition to steam generator resulting in overcooling of the RCS. (SIR S02-82-260 Reference 12)	Visual	No indications of damage to piping or supports.

NOTES:

(1) Hands On means an in-place rotation of mechanical snubbers about their axis looking for hesitation or binding.

Table 2

Inspections of Snubbers on Main Feed Line FW-189

Date	Reason for Inspection	Type	Findings
December 1982	50% Power, Thermal Expansion Test	o Visual o Manual Stroking	o NCR 2-032 S2-FW-189-H013 Top-extension arm buckled S2-FW-189-H013 Bottom-loose extension nut o NCR 2-068 S2-FW-189-H010 both-jammed o NCR 2-069 S2-FW-189-H017 both-jammed All of the above snubbers, exept serial number 4322 on support S2-FW-189-H010 were replaced (2)

NOTES:

2) Serial number 4322 was manually stroked with satisfactory results and reinstalled.

thought to have failed. However, after successful manual stroking it was reinstalled. In his review of the December 1982 failures, Mr. Mertens requested that this snubber be removed and sent to the manufacturer for a functional test under load and an inspection of the internals. This snubber was found to have no internal damage and to be operational, a confirmation of the manual stroke test in the field. Note, that this snubber was one of two previously replaced in January 1982 and hence should not have damaged at the time of the functional load testing. Thus, with the appropriate inspection technique, snubber inoperability has been detected in the field without the use of functional load testing.

Since the manufacturer's test program conducted by Mr. Mertens has shown that all failure modes cannot be detected by manual stroking, additional test measures will be required. Further discussion of this is provided in response to item 4.

Item 3: "An attempt was made to obstruct the engineering evaluation I conducted."

Reply: This appears to relate to difficulty experienced in March, 1983, in removing the snubber (S/N 4322) installed in January, 1982. As indicated, Mr. Mertens' supervisor supported this effort, which was successful when the appropriate documentation (a NCR) was prepared. As discussed above, this snubber was found to be undamaged and operable.

Item 4: "The method presently used for inspecting snubbers is inadequate and misleading."

Reply: The inspection program for mechanical snubbers is defined in the SO23 Technical Specifications Section 4.7.6 and the ASME Boiler and Pressure Vessel Code, Section XI. The Technical Specifications require periodic visual and functional inspections. The visual inspection acceptance criteria verify that:

1) there are no visible indications of damage or impaired operability, and;

2) attachments to the foundation or supporting structure are secure.

The functional test acceptance criteria verify that:

1) activation (restraining action) is achieved within the specified range in both tension and compression, except that inertia dependent, acceleration limiting mechanical snubbers may be tested to verify only that activation takes place in both directions of travel.

2) Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range.

Item 4: (Continued)

3) Where required, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel.

4) For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

5) Fasteners for attachment of the snubber to the component and to the snubber anchorage are secure.

Additionally, the Technical Specifications require that a special inspection shall be performed each refueling outage of all snubbers attached to sections of safety related piping systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and visual inspection of the systems. This inspection shall satisfy the requirements of the visual acceptance criteria and confirm freedom of motion of the snubber.

An additional inspection method used for mechanical type snubbers in the field is to stroke the snubbers through their range of travel, looking for any binding of the unit. This method is consistent with the recommendations of the snubber manufacturer and meets the requirements of the special Technical Specification inspection mentioned above.

Inservice inspection requirements for snubbers at Units 2 and 3 are contained in Section XI, ASME B&PVC, 1977 Edition and addenda through Summer 1979. Visual and functional testing acceptance criteria contained therein are consistent with those specified in the Technical Specifications.

As was noted earlier in responding to Item 1, the visual method will not disclose all types of snubber failure modes. It is for this reason that manual stroking was performed in December 1982. During that inspection, the manual stroking method was an effective, discriminating method for determining snubber operability. Since the testing performed under Mr. Mertens' cognizance at Pacific Scientific demonstrated that visual inspection and manual stroking cannot detect all failure modes, the choice of inspection method should consider functional load testing. An onsite bench testing machine has been on order for some time and delivery is currently forecast for October 1983. Should testing be required before October, portable onsite or offsite testing services are available in a reasonable time period (approximately two days).

Thus, there are several levels of mechanical snubber inspection available. Should operating conditions develop which are known to produce significant dynamic transients, a Station Incident Report would be prepared and an engineering evaluation performed. The scope of this evaluation would include testing requirements for mechanical snubbers and selection of the most appropriate inspection technique.

Item 5: "No effective QA program exists for monitoring snubber performance as per Technical Specification page B 3/4 7-6, Snubber Basis."

Reply: Page B 3/4 7-6 of the Technical Specifications discusses the requirements for monitoring the service life of snubbers subject to environmental degradation (i.e., hydraulic snubbers).

Presently, an effective program, implemented by procedures, is in place to monitor both hydraulic and mechanical snubbers. This program consists of both a visual and functional test to ensure proper snubber performance. A computer-based system which records the maintenance history of all snubbers is utilized to input information into the snubber surveillance program. Any failures that are identified during these inspections are subjected to a detailed engineering analysis and, if determined necessary, an increase in the frequency of inspection or the number of snubbers inspected is implemented.

Item 6: "A recommendation to install minimum instrumentation for monitoring operability of snubbers in safety-related piping was dismissed."

Reply: Mr. Mertens participated in a meeting between representatives of NUS Corporation, the Electric Power Research Institute (EPRI) and SCE in early June 1983, which included discussion of this recommendation. Pursuant to that meeting, SCE expects a proposal from NUS in early August, 1983, which will include the posssibility of use of monitoring instrumentation.

Item 7: "The engineering evaluation performed by Bechtel Power Corporation to determine the cause of failure is inadequate and misleading."

Reply: The primary purpose of the engineering evaluation performed by Bechtel was to determine if allowable piping stress limits had been exceeded and the extent of inspection and repair required to return the line to service. The analysis which is sumarized later in this section, clearly indicates acceptable stress levels and provides appropriate direction for inspection.

Since the engineering failure analysis showed that stress levels were acceptable, the replacement of all failed snubbers on the main feedwater line FW-189 and a review of the subsequent thermal expansion data indicated that the line was moving as predicted by original design calculations. After replacement, snubbers at support S2-FW-189-H-013 were measured to extend 1.90 inches in the north direction at 50% power. This compares almost exactly with analytically calculated movement of 1.85 inches in the north direction.

SCE requested BPC on April 7, 1983, to conduct a further evaluation as to the specific causes of the snubber failure and to determine any long term corrective action. This evaluation is scheduled to be complete by August 19, 1983.

Item 8: "Snubbers on the main feedwater lines inside the Containment were not designed for dynamic loads, they are undersized and may not be able to ensure structural integrity of the main feedwater lines."

Reply: Main feedwater lines inside the Containment, including their supports, are designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class 2. The design has been verified as correct. As indicated in the design basis (FSAR paragraph 10.4.7.1.d), it does not include dynamic loading resulting from severe waterhammer transients, as these are to be avoided by operational and design measures. The startup test on March 21, 1981, which is hypothesized to have caused the damage to the five snubbers which was revealed by the inspections of December, 1982, and to the two snubbers which were replaced in January, 1982, was substantially more severe than is included within the design basis. Further, following steam generator feedring modifications in mid-1981, a revised startup test was successfully conducted in June 1982 which confirmed the adequacy of the main feed system design basis. This information was provided to the NRC at their request in Responses to Questions 010.18 and 010.54 (Attachments 5 and 6). Further evaluation of the potential for damaging waterhammers in the feedwater piping was performed in 1982 in response to requests from Station and Startup Engineering. Attachments 7, 8, 9 and 10 provide copies of these requests and the evaluations performed.

In summary, the snubber damage probably occurred during startup testing, which included a severe waterhammer transient. Extensive modifications, testing and evaluations have been performed to reduce the potential for damaging waterhammer events and to show that the feedwater piping will not be overstressed. Should such a transient beyond the design basis recur during plant operation, careful inspection and testing of affected snubbers would be performed in accordance with existing Station procedures including consideration of this startup testing experience. As discussed in Section IV of this report, criteria for determining when such a transient might have occurred will be developed.

Item 9: "An attempt was made to intimidate me [Mr. Mertens] (transferred or fired) in response to my questioning BPC's performance at SONGS 2 and 3." [The comments within brackets have been added for clarification.]

Reply: No attempt was made to intimidate Mr. Mertens. In an exchange of correspondence between Mr. Mertens, SCE and others commencing in November, 1982, he requested:

- o Total expenditure data concerning SONGS 2 and 3
- o Payments to Bechtel for engineering work on SONGS 2 and 3
- o A copy of the contract between SCE and Bechtel for SONGS 2 and 3
- o The same information as above for SONGS 1

Mr. Mertens pursued these documents and data as a stockholder with the Secretary's office, rather than as an employee. In his letters, he included the comment that the reason for his request ". . . is to determine whether or not the [contract] contains provisions for redress in the event of unsatisfactory performance, and to study the terms of such provisions." At no time prior to receipt of his letter to Mr. Gould dated July 20, 1983, were questions of nuclear safety raised in this correspondence.

Item 9: (Continued)

Mr. Mertens was counseled in February, 1983, to stop his repeated attempts to obtain the information above, as SCE has declined to provide it. This counseling did not adversely affect his overall performance rating in his March, 1983, performance appraisal, and he was not threatened with transfer or dismissal.

Item 10: "The original and all copies of NCR S01-P-1303, Rev. 1, were destroyed."

Reply: Revision 0 of this NCR for SONGS 1 was opened on December 8, 1982. The condition described relates to corrosion of a heat exchanger foundation and is also described in an NRC inspection report transmitted to SCE by NRC letter dated January 7, 1983. The work required to repair the foundation is ongoing and is about 80% complete. Revision 1 of the NCR was drafted to readdress the original evaluation of heat exchanger operability which had been made for Revision 0. Some difficulty was experienced in deciding how to treat operability during the repair. It was concluded in May 1983, prior to validating the NCR revision that readdressing operability was not necessary, and the repair could be made with the heat exchanger in service, so the revision was not completed. Revision 0 of the NCR remains in effect, and it governs the work which is now nearing completion.

In addition to the ten concerns identified above, additional items were presented in Mr. Mertens' memorandum to Mr. B. Katz dated May 23, 1983, (Attachment 1). Each is listed verbatim below along with a reply.

Item 1: "The pressure differential of 500 to 800 psi (see Attachment [A], Item 6, Page 2) responsible for collapsing the steam generator feedring induced a pressure wave back through the water-filled main feedwater line producing, at the instant of reflection from the 20" check valve flapper, the dynamic force that damaged the snubbers at locations H010, H013 and H017." [See Attachment 1, Paragraph B, Conclusion 2.]

Item 2: "G-Forces produced by the respective transient may have exceeded design limits. (An inspection of main and auxiliary feedwater lines for signs of overstressing is advised)." [See Attachment 1, Paragraph B, Conclusions 2 and 4.]

Reply to Items 1 & 2: The postulated differential pressure and resulting G-forces produced by the transient did not result in piping stresses that exceeded design limits. This conclusion was reached through inspection and analysis performed immediately following the discovery of damaged snubbers.

Inspection

Visual inspection of the snubbers and associated snubber support structures identified only snubber component damage, and verified that no damage was evident on associated support structures.

Analysis

Four analyses were performed to evaluate stress levels in the piping.

A. Limit Load Analysis

A limit load is the calculated load at which the weakest part of the snubber support structure would fail. This failure would be evidenced by physical deformation of structural members or cracked welds. A limit load of approximately 56,000 lbs. was calculated at support H015. This limit load was then applied to the feedwater piping system to determine resulting stress in the piping.

B. Check Valve Slam

Forces up to 50,000 lbs. were calculated to occur due to a pressure transient. These forces were then applied along the axis of the pipe to determine resulting stress levels.

C. Steam Generator Nozzle Movement

The steam generator nozzle was analytically moved to simulate thermal growth of the NSSS and steam generator. At the same time snubbers at support location H013 were modeled in a locked condition in order to simulate r sistance to feedwater pipe movement.

D. Cold Pull

The piping was analytically "cold pulled" to simulate the effects of thermal growth against locked snubbers. This additional analysis was performed after the remaining snubbers at support locations H010 and H017 were found to be damaged late in December 1932.

The calculated results of these analyses indicated low stress levels in the piping:

1. Local stresses at support attachment locations were approximately 15,000 psi compared to an allowable yield stress of 32,000 psi at 400°F.

2. Piping component primary stresses were approximately 18,000 psi. This is also below allowable stress of 21,000 psi.

3. Piping component secondary plus primary stresses were maximum of 34,000 psi due to consideration of steam generator nozzle thermal growth and piping thermal growth against locked snubbers at H010, H013 and H017. This compares with an allowable secondary stress of 43,750 psi.

Item 3 : "The engineering evaluation of the damaged snubbers on FW-189 Unit 2, performed by BPC (Attachment A, Item 4) made no attempt to examine the snubbers internally to determine the true magnitude of the shockforce; the recommendation for larger snubbers should not have been cancelled without further study, and it should not have advised SCE in a follow-up letter (Attachment A, Item 5) that no further investigation of the snubber failure was recommended." [See Attachment 1, Paragraph C, Finding 5.]

Reply: An inspection of the snubbers on feedwater line FW-189 was directed by Bechtel Engineering in December 1982, as a result of monitoring piping movements as required by Test Procedure 2PA-102-01. The subsequent inspections at support locations H010, H012, H013, H014, and H017 resulted in the replacement of five damaged snubbers.

The engineering evaluation recommended that the damaged snubbers be inspected and replaced if necessary. SCE Engineering made arrangements to send these snubbers to Pacific Scientific for failure analysis as part of the continuing evaluation program.

The recommendation that no further evaluation of the snubber failure was required was based upon the determination that acceptable piping stress limits were not exceeded and future events that could cause similar snubber damage would require a detailed inspection per the plant Technical Specifications. Further evaluation as to the mechanism that imposed abnormal loads on the snubbers is scheduled to be complete by August 19, 1983. The magnitude of shock force was determined as discussed in Item 4 below.

Item 4: "Magnitude of Dynamic Forces"

The magnitude of the shock forces that acted on the snubbers can be estimated from the severity of the damage their internal parts suffered.

An examination of the five damaged snubbers at the manufacturer's test facility revealed that all failed under compression. This fact tells us the direction of the force and its point of origin. With the aid of Figure (a) on Drawing No. 100, this is explained as follows: Figure (a) shows vector diagrams corresponding to compressive forces acting on the snubbers. It can be seen that the force F1 must have been the result of a check valve slam and that component forces F2, F3 and F4 caused the snubbers to fail.

From the extent to which the snubber internals were damaged, one can establish the approximate value of the destructive force. On all of the five snubbers the threads between the ball screw shaft and the inner thrust bearing race (Figure 1) were sheared, and further, all shaft ends had hit bottom and suffered plastic deformation.

Shearing the threads requires a static force of approximately 30,000 lbs. as confirmed by a test conducted at the manufacturer's test lab (see test report Attachment A, Item 2).

Item 4: (Continued)

Examination of Figure 1 reveals that, after having sheared the threads, the shaft was free to travel another 3/16" before hitting bottom and becoming deformed. Therefore, shear and deformation forces are additive. The transient force causing the deformation is estimated as follows:

The shaft material is ASTM A322, having a yield stress of 75,000 psi. The shaft end area is 0.2 inches squared. From these two quantities, we obtain 15,000 lbs. as the force of deformation. Adding the two forces yields 45,000 lbs. as the approximate minimum force acting on one snubber. Since the snubbers are arranged in pairs, we obtain, for the smallest component force, F2 = 90,000 lbs. (Figure (a). For establishing a minimum value for the resultant F1, the following reasoning was applied: Vector diagrams in Figures (a) and (b) on Drawing No. 100 show component forces F2 and F4 to be the smaller ones. If 90,000 lbs. is assigned to both F2 and F4, the primary force F1, acting on the check valve plate, becomes F1 = 170,000 lbs. and component force F3 = 140,000 lbs. These are minimum forces because the extent of damage inflicted upon the feedwater pipe snubbers, when compared to the degree of damage seen in the snubber subjected to the dynamic destruct test, leaves no doubt that greater forces were at play." [See Attachment 1, Paragraph E, Analysis 1.]

4. Reply: The damaged snubbers at support locations H010, H013 and H017 are PSA-10's which are capable of reacting approximately 32,000 lbs. before failure. These loads were developed by Pacific Scientific thru testing. A review of the structural members that transmit the snubber loads into the building structure reveals that the highest loads that could have occurred at support locations H010, H013 and H017 are approximately 66,000, 56,000, and 62,000 lbs., respectively. Since there was no visible physical evidence of damage to these pipe support structures, these loads are considered to be upper bound. Application of these loads to the piping system has been performed to confirm the acceptability of stress levels on the piping as detailed in the reply to Items 1 and 2.

Detailed inspections of the piping system and associated pipe supports including snubbers indicated no failures of the structural steel members or connecting welds. Therefore, loads on the order of 90,000 lbs. and 140,000 lbs. could not have developed since structural steel member failures would have occurred at significantly lower levels (66,000, 56,000 and 62,000 lbs.).

Item 5: Shock Arrestor Size

It has been shown that each snubber was exposed to a shock force of at least 45,000 lbs. They are designed for 15,000 lbs. Since conditions exist that could cause another waterhammer in FW-189, it is advisable to increase the size of snubbers on the main feedwater lines of Units 2 and 3. This may require redesign of the components to which the snubbers are attached." [See Attachment 1, Paragraph E, Analysis 2.]

Reply: Our evaluation does not concur that shear and deformation forces are additive and that each snubber was exposed to a shock force of at least 45,000 lbs. We believe that a load of approximately 32,000 lbs. to 37,000 lbs. could have been exerted on each snubber.

The destructive tests conducted by Pacific Scientific on a number 10 snubber indicated a failure load of 31,600 lbs. and dynamic loads up to 37,000 lbs. This load resulted in sheared threads between the ball screw shaft and the inner thrust bearing race. Subsequent to shearing of the threads the shaft must travel 3/16" inches before coming into contact with the snubber housing. Therefore, the loads generated to cause shearing of the threads did not occur at the same time as the load causing shaft end deformation. Therefore, the shear and deformation loads are not additive. The maximum load therefore, is the larger of the two loads, i.e. 31,600 lbs. to 37,000 lbs.

The design of the backup structure for the paired snubber arrangement is such that it will sustain visible damage for loads of approximately 60,000 lbs. Since there was no visible physical damage on the backup structure, it is not credible to postulate a failure load of 90,000 lbs. (a combined load of 45,000 lbs. on each of two snubbers).

Item 6: "Inconsistency between Existing Snubber Design Loading and Technical Specification.

An inconsistency exists between the intent of tne T/S and the design basis for snubbers on the safety-related main feedwater system, as the following explains.

T/S 3/4.7.6 Snubbers, Plant System Bases reads as follows:

PLANT SYSTEMS

BASES

3/4.7.6 SNUBBERS

6. Reply: There is no inconsistency between the design basis for the feedwater piping system and Technical Specification 3/4.7.6. FSAR Paragraph 10.4.7.1.D states "Design considerations of the feedwater piping and steam generator feedring preclude the occurrence of hyraulic instabilities." Accordingly, the main feedwater line snubbers were not designed for abnormal hydraulic dynamic loads, but were designed for Seismic Category I loads. Unusual events such as the feedring integrity test of March 1981 or the cooldown of November 1982 were analyzed in detail. Procedures and design changes have minimized the possibility of recurrence. Additionally, analysis showed that these events did not result in excessive pipe stress.

Other scenarios will be evaluated as part of the continuing snubber failure analysis which is scheduled to be complete by August 19, 1983.

All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant System and all other safety-related systems is maintained during and following a seismic <u>or</u> other <u>event initiating dynamic loads</u>. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure, or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

The main feedwater line snubbers, however, were never designated for dynamic loads (see Attachment A, Item 10, DYN. LDS column).

Since conditions that can cause a waterhammer in the main feedwater line still exist, (1) the event can repeat itself, causing the snubbers to fail again.

Therefore, the snubbers, as they are, cannot ensure the structural integrity of the system during and following a dynamic load event. This requires resolutions in regard to operability determinations for the feedwater system."
[See Attachment 1, paragraph E, Analysis 4.]

Item 7: "QA as related to snubbers is inadequate" (see Attachment 1, Proposed letter to Mr. J. B. Martin, page 4).

Reply: In the July 20, 1983 proposed letter to Martin, Attachment A, Item 7 is a copy of SCE Nonconformance Report (NCR) S023 F-463, dated 1/25/82, and identifies that snubber #2609 cannot be rotated about its axis and that cold setting information on ID tags of both snubbers #2609 and #2603 do not match the design drawings.

In the July 20, 1983 proposed letter to Martin, Attachment A, Item 8 is a copy of work order 2113 which identifies a snubber as being incorrectly tagged and indicates that a new tag was affixed.

The inference of inadequate Quality Assurance, regarding Attachment A, Items 7 and 8, appears to be based upon a comparison of the identification of an inoperable snubber (#2609), including the disposition requiring its rejection, quarantine and subsequent replacement, with work order 2113, requiring only replacement of the ID tag. This apparent inadequacy is further alluded to on page 4, first paragraph, of the letter proposed to be submitted to Mr. J. B. Martin, NRC Regional Administrator, Region V.

It must be noted that verification of the removal, quarantine and replacement of Snubber #2609 is evidenced in work order 2114, not 2113 as specified. This work order (2114) is additionally referenced on NCR S023 F-463 in Block 24 (Reference 8).

Based upon the proper research of NCR S023 F-463 and verification documents utilizing both work orders, there is no validity to the comment inferring inadequate Quality Assurance.

IV. PRESENTATION OF CONCERNS TO ONSITE REVIEW COMMITTEE

In accordance with the SCE procedure for the review of nuclear safety concerns (see Attachment 2), a special Units 1, 2 and 3 Onsite Review Committee meeting was held on July 28, 1983. Mr. Mertens presented four concerns related to nuclear safety. These concerns (taken from the minutes, Reference 1) are listed below:

(1) Visual inspection methods for mechanical snubbers do not tell us anything about operability of the snubbers.

(2) There are no means of detecting transients that c[illegible] damage snubbers.

(3) There are no means of detecting damage to snubbers except for gross deformations.

(4) Piping (specifically, FW189) may be overstressed as a result of the damaging transient. (Concern with respect to power operation in the next week.)

Item 4 was declared a restraint to Mode 2 entry for Unit 2 until a review of the feedwater line FW189 stress analysis was completed. The results of that analysis are contained in Section III of this report. The analysis concludes the line was not overstressed.

A second special Onsite Review Committee meeting was held August 4, 1983 (Reference 2) to complete the review of items 1, 2 and 3. The Committee concluded that potentially damaging dynamic transients can be detected by observation of operational parameters. Additionally, the Committee concluded that visual inspection is not capable of detecting all modes of mechanical snubber failure. Thus, the Committee directed that criteria shall be established, in administrative procedures, for performing functional mechanical tests of snubbers. This criteria will be developed based on the conclusion that transients which will induce mechanical snubber failure will be events that are known and that current procedures lack a definition for events that should trigger functional testing of mechanical snubbers in selected areas based on the possibility that a potentially damaging waterhammer could have occurred.

V. SUMMARY OF ACTIONS

NOT COMPLETE AT TIME OF THIS REPORT

1. From Section III, further evaluation as to the mechanism that imposed abnormal loads on the snubbers is scheduled to be complete by August 19, 1983.

2. From Section III, other scenarios will be evaluated as part of the continuing snubber failure analysis which is scheduled to be complete by August 19, 1983.

3. From Section III, further evaluation as to the specific causes of the snubber failure and to determine any long-term corrective action.

4. From Section IV, the Onsite Review Committee has directed that criteria for determining when potentially damaging waterhammers might have occurred will be developed by August 30, 1983.

ATTACHMENT 1

(1)

Mr. W. R. Gould
Chairman of the Board
Southern California Edison Company
P.O. Box 400, 2244 Walnut Grove Avenue
Rosemead, California 91770

RECEIVED
JUL 22 1983
WM. R. GOULD

July 20, 1983

Dear Mr. Gould:

I am writing to you because I must assume that you are not being fully informed of certain matters concerning nuclear operations.

I am a station engineer at Songs and I have enclosed for your review two letters that I have written; one to Mr. J. B. Martin, NRC, and the other to Mr. L. Grimes, Chairman, CPUC.

I believe Company policy reqires employees to inform management of such correspondence and submit the same for review. The explanation for writing them can be found in their text.

It is my intention to mail the letters on July 27, 1983. I would appreciate receiving your reply as to whether or not they contain false statements so that I may enclose it.

Sincerely

John Mertens

John Mertens

Enclosures

3109 S. El Camino Real · San Clemente, CA 92672 · (714) 498-6311