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ACRS MEETING MINUTES ON THE MEETING OF THE
WESTINGHOUSE WATER REACTORS SUBCOMMITTEE
FEBRUARY 12, 1986
WASHINGTON, DC

Purpose

The ACRS Subcommittee on Westinghouse Water Reactors met on February 12, 1986, at 1717 H Street, NW, Washington, DC. The purpose of this meeting was to discuss the November 21, 1985 water hammer event at SONGS-1. The discussions were primarily focused on the Incident Investigation Team (IIT) event investigation report. The Subcommittee observations at this February 12, 1986 meeting are to be discussed during the February 13-15, 1986 ACRS meeting. Action by the ACRS is currently expected to be needed during the April-May 1986 timeframe. It is expected that the actions to be taken by SONGS-1 and the generic implications of the event will have been developed by this time. The SONGS-1 restart is currently scheduled for May 20, 1986.

The February 12, 1986 Subcommittee met between 8:30 a.m. and 12:30 a.m. and heard presentations from representatives of NRR, the IIT, and Southern California Edison (SCE). The entire meeting was open to the public. The principle attendees were:

ACRS

J. Ebersole, Subcommittee Chairman
H. Etherington, Member
C. Michelson, Member
C. P. Siess, Member
G. Reed, Member
D. Ward, Member
W. Kerr, Member
I. Catton, Consultant
R. Savio, Staff

NRC

W. Lanning, AEOD

J. Heltemes, AEOD

R. Hernan, PPA

R. Dudley, NRR

T. Martin, Region 1

Southern California Edison

B. Duncil

W. Baskin

J. Rainberry

M. Medford

Highlights

1. The San Onofre Nuclear Generating Station, Unit 1, operated by the Southern California Edison Company (SCE), is a 450 MWe Westinghouse pressurized water reactor located on the Pacific Ocean, approximately four miles south of San Clemente, California. The plant received an NRC operating license in 1967. The plant had been shut down for refueling on February 27, 1982. The SEP review which was being carried out at this time raised concerns as to the plant's capability to meet the original seismic design basis. As the results of discussions with the NRC, SCE agreed to make modification which would result in upgrading the seismic design of the plant. A subsequent August 11, 1982 NRC order precluded restart of the plant until the seismic concerns were resolved. The plant was eventually restarted in November 1984.

   The plant was operating at 60 percent power on November 21, 1985 and the operators were attempting to locate a ground fault in a transformer which was supplying power to one of two-safety-related 4160V electrical buses. The resulting isolation of the transformer caused the safety-related bus to de-energize, which tripped all

feedwater and condensate pumps on the east side of the plant. The pumps on the west side of the plant were unaffected since their power was supplied from another bus. The continued operation of the west feedwater and condensate pumps, in combination with the failure of the east feedwater pump discharge check valve to close, resulted in over-pressurization and rupture of an east-side flash evaporator low-pressure heater unit. The operators, as required by the emergency procedures, tripped the reactor and turbine-generator.

As expected, this resulted in a loss of plant AC power. The subsequent 4-minute loss of inplant electric power caused emergency diesel generators to start (which by design did not load), de-energized all safety-related pumps and motors, significantly reduced the number of control room instrument indications available for operators to diagnose plant conditions, and produced spurious indications of safety injection system actuation (which in turn caused the NRC red phone on the operator's desk to ring).

The loss of steam generator feedwater was the result of the loss of power to the two main feedwater and one auxiliary feedwater pump motors, and the, as designed, 3-minute delay in the startup of the steam-powered auxiliary feedwater pump. The loss of the feedwater pumps, in combination with the failure of four additional feedwater check valves to close, allowed the loss of inventory from all three steam generators and the partial voiding of the long horizontal runs of feedwater piping within the containment building. The subsequent automatic start of feedwater injection by the steam-powered auxiliary feedwater pump did not result in the recovery of steam generator level because the backflow of steam and water to the leak in the evaporator carried auxiliary feedwater with it. The operators isolated the feedwater lines from the steam generators, as required by procedure, unknowingly initiating the process of refilling the feedwater lines in the containment building.

Before all feedwater lines were refilled, a severe water hammer occurred that bent and cracked one feedwater pipe in the containment building, damaged its associated pipe supports and snubbers, broke a feedwater control valve actuator yoke, stretched the studs, lifted the bonnet, and blew the gasket from a 4-inch feedwater check valve. The damaged check valve bonnet was the cause of a leak.

2. As a result of the November 21, 1985 event, an Incident Investigation Team (IIT) was formed by the NRC and dispatched to the site. The team members were T. Martin, Team Leader (Region I), M. Chiramal (AEDO), W. Kennedy (PSRB), W. Lanning (AEOD), A. Serkiz (GIB), and S. Showe (IE). The Team subsequently issued a report (NUREG-1190, January 1986) and reported their findings to the Commission on January 20, 1986. The IIT findings are summarized in Attachment A. NRC Staff assignments as to followup actions from the IIT investigation were made via a February 4, 1986 memo from V. Stello. A copy is enclosed as Attachment B. Plans and schedules for the completion of these items are expected by March 1, 1986.

3. Thomas Martin summarized the IIT investigation report. The most significant problem uncovered in the event was the undetected failure of the five check valves. It is likely that this will lead to a significant generic action on the part of the NRC. Operator response in the management of the November 21 event was judged to be good. Communications with the NRC and the NRC emergency response were not adequate and indicate the need for better planning and training of both SCE and NRC personnel. A number of equipment malfunctions were identified and are summarized in Attachment C. The root causes of the check valve failures have not been determined and are still under review by SCE and the NRC Staff. The IIT believes that the potential contributors to this event were inadequate maintenance, inadequate inservice testing, inadequate

design, and inadequate consideration of the effects of reduced power operation on check-valve operation. It was noted that the failed check valves and check valves of similar design have had a history of failures. The failed check valves were "oversized" for the application and this contributed to their susceptibility to failure at reduced flow. (The lift forces at reduced flow were not large enough to lift the valve disk out of the flow stream. The valve disks were subjected to an increased level of flow-induced vibration when in the flow stream.) The IIT was not able to reconstruct the history of past check valve failures at SONGS-1 from the information in the maintenance records. The inservice testing procedures for this check valves were judged to be inadequate in detecting check valve degradation.

4. SCE intends to replace the check valves of the type involved in the check valve failures with a different design. The old (Pacific Valve) valve design and the replacement (Atwood-Morrill) valves are shown on Figure 1. The configuration of the feedwater lines will be modified as to reduce the potential for water hammer. A number of other modifications are being proposed by SCE. These modifications are summarized on Attachment D.

5. There was discussion as to the event and the generic implications. It is expected that the Subcommittee will meet to discuss this subject again in the near future. Some of the issues raised by the Subcommittee during these February 12 discussions were as follows.

   (a) Water hammer appears to be still a significant problem and should be addressed seriously by both the NRC Staff and the Industry and previous conclusions as to design measures to prevent water hammer should be reexamined.

(b) The role of the NRC and the use of the ENS should be reevaluated.

(c) Existing AFWS reliability studies should be reexamined to determine if water hammer has been properly considered.

(d) Plants which do not have the capability to remove decay heat via a feed/bleed process (for example, CE System 80 plants) are more dependent on feedwater systems and should be examined in this light).

(e) Present conclusions as to the vulnerability of plants to interface LOCA's should be reexamined in the context of the multiple check valve failures at SONGS-1

(f) Placement of check valves downstream of flow control valves should be reviewed at SONGS-1. The flow control valves will cause downstream turbulence which could inturn cause flow-induced vibration of the check valve disks.

(g) It is important for plant management to get feedback from operations and maintenance personnel as to the adequacy of the plant design and to use this information in making the necessary plant changes.

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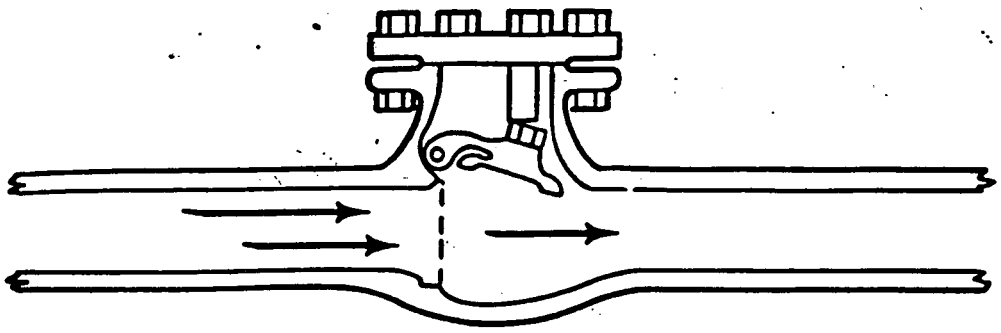
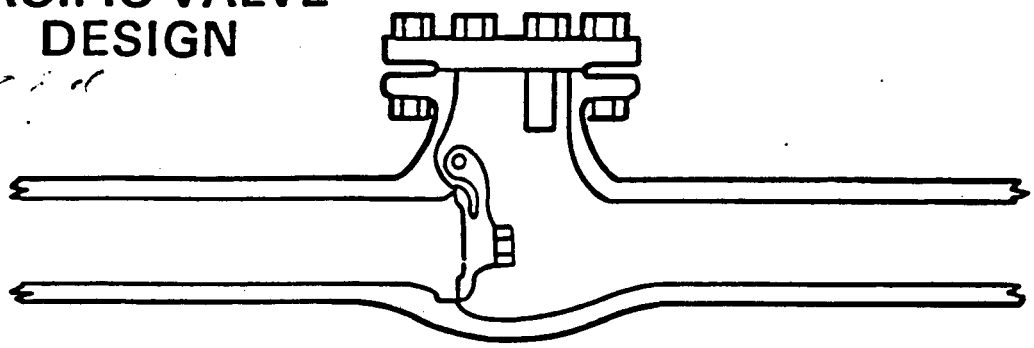
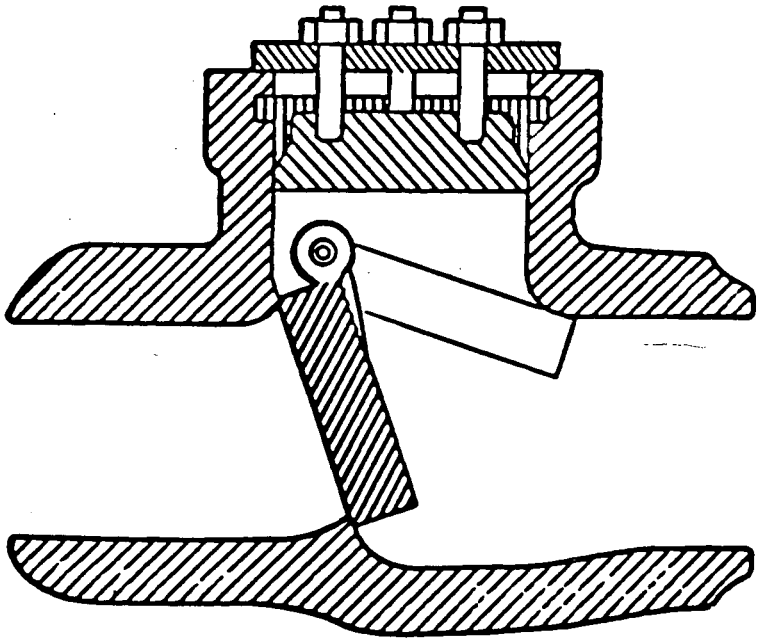
NOTE: Additional meeting details can be obtained from a transcript of this meeting available in the NRC Public Document Room, 1717 H Street, NW, Washington, DC, or can be purchased from ACE-Federal Reporters, 444 North Capitol Street, Washington, DC 20001, (202) 347-3700.

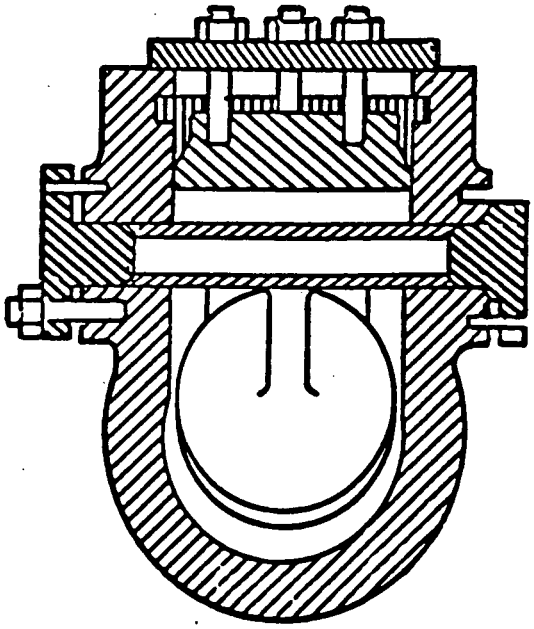
FIG 1



PACIFIC VALVE DESIGN







ATWOOD-MORRILL VALVE DESIGN

ATT A

9 PRINCIPAL FINDINGS AND CONCLUSIONS

The event that occurred at San Onofre, Unit 1 on November 21, 1985, was significant because (a) all inplant ac power was lost for 4 minutes; (b) all steam generator feedwater was lost for 3 minutes; (c) a severe water hammer was experienced in the feedwater system which caused a leak, damaged plant equipment and challenged the integrity of the plant's ultimate heat sink; (d) all indicated steam generator water levels dropped below scale; and (e) the reactor coolant system experienced an acceptable but unnecessary cooldown transient. In addition, other aspects which contributed to the complexity of the event and to the burden placed on the operators included: a rupture in a flash evaporator unit; spurious and incomplete instrumentation indications; fire alarms and system actuations; and malfunctions in the automated security equipment.

The Team has concluded that the most significant aspect of the event was that five safety-related feedwater system check valves degraded to the point of inoperability during a period of less than a year, without detection, and that their failure jeopardized the integrity of safety-related feedwater piping.

The root causes of the check valve failures have not been determined and are still under review by SCE and its contractors. Potential contributors to this problem include inadequate maintenance, inadequate inservice testing, inadequate design, and inadequate consideration of the effects of reduced power operations. Maintenance records for these valves were either missing or lacked specificity on what was done. Inservice testing records for these valves were inconsistent; the testing procedure was not rigorous; the test acceptance criteria were subjective; the testing frequency was open-ended; and, the tests did not assure detection of the failures found. These check valves and valves of similar design have a history of like failures. Finally, reduced power operations at Unit 1 are now routine because of steam generator tube plugging and sleeving, and the reduced feedwater flow may have increased the susceptibility of check valve components to hydraulic-induced vibration.

In addition to this major conclusion on the underlying cause of the event, the Team has made the following related findings and conclusions. There is no significance to the order in which they are presented.

1. The primary cause for the water hammer in the feedwater piping was the failure of multiple check valves in the feedwater system. These failures permitted the piping to empty and fill with steam before the motor-operated feedwater isolation valves were closed. Although the steam condensation-induced water hammer occurred in only one feedwater line, the potential existed for water hammer to occur throughout the safety-related portions of the feedwater system.

2. The failures of the five check valves in the feedwater system provided a mechanism for potential common mode failure of the heat sink provided by the three steam generators. The failed check valves permitted high pressure steam and water from the steam generators to flow back to the low pressure condensate system; the backflow carried with it the auxiliary

feedwater flow necessary to maintain the heat sink provided by the steam generators. Operator actions were necessary to stop the backleakage and prevent a more serious sequence of events.

3. Long horizontal runs of feedwater piping with the potential for voiding are particularly susceptible to destructive steam condensation-induced water hammers. Further, operators are not provided the means for detecting the voiding of these lines or given guidance on appropriate ways to deal with the situation. Design or procedural changes may be warranted.

4. The flash evaporator failed when overpressurized by the discharge flow of an operating feedwater pump due to the partial loss of power and a stuck open feedwater pump discharge check valve that should have prevented the backflow.

5. The timing of the five check valve failures could not be ascertained with certainty. The Team concluded that all check valves had failed prior to the event because the missing parts to the valves were not found in the inspected feedwater piping after the event. Noise from the B steam generator feedwater piping, evident to plant personnel since June 24, 1985, supports the conclusion that the feedwater control station check valve in the B feedwater line had failed earlier. The inspection of the steam generators has not yet been completed by SCE.

6. The surveillance procedure for testing the check valves in the Inservice Testing (IST) program lacked adequate methods and objective acceptance criteria for determining whether check valves are closed. Thus, although the check valves had been tested within the past year, operators may have misinterpreted the test results. Furthermore, the IST is not designed to detect developing conditions that may lead to the failure of the check valves, e.g., loose disks and stud nuts.

7. The NRC had not completed its review of SCE's Inservice Testing Program. The initial program was submitted in September 1977 and revised in its entirety on January 24, 1984. Disagreement between SCE and NRC on resolution of certain open issues and scheduling problems with NRC's review have substantively contributed to this delay.

8. The resolution of the Unresolved Safety Issue, USI A-1, "Water Hammer," did not specifically address the prevention and mitigation of the consequences of condensation-induced water hammers in feedwater piping upstream of the feedring. Interviews of NRC staff involved in resolution of water hammer issues failed to develop citable references, decisions, or discussions that provided a basis for excluding further consideration of feedwater piping water hammer. However, in the regulatory analysis of the resolution of USI A-1, the staff acknowledged that elimination of water hammers is not feasible, that the frequency of water hammers had been substantially reduced by changes in design and operations, and that studies of water hammer had revealed a significantly lesser safety concern than previously hypothesized. It appears that further consideration of water hammers due to main feedwater line voiding was not pursued due to a lack of reported occurrences in U.S. plants.

9. NRC's reliance on "J" tubes to delay the development of conditions necessary to support steam generator water hammer implicitly assumes that feedwater check valve integrity would be maintained to prevent steam generator

feedring voiding. However, corresponding regulatory requirements to ensure that these check valves performed this safety function were not part of the resolution of the water hammer issue.

10. The root cause for the loss of power was a phase-to-phase fault of an electrical cable from auxiliary transformer C to bus 1C. The underlying reason for the cable failure has not yet been determined; however, it appears that the cable may have become wetted by a long-term flange leak from the feedwater system, running above the cable tray.

11. The plant is designed to experience an extended loss of inplant ac power on loss of offsite power without safety injection. Operators are required to restore power from the switchyard or to load the diesel generators to restore inplant power. SCE's Emergency Operating Instructions on loss of ac power lack guidance on how long operators can attempt to restore power from offsite sources before the diesel generators should be loaded following a loss of inplant ac power, or how long the diesel generators can run unloaded without overheating, if their ac-powered radiator fans remain de-energized.

12. The station loss of voltage auto transfer scheme for establishing the delayed access to offsite power may not have functioned as designed. SCE evaluations are continuing.

13. The multiple spurious indications early in the event that a safety injection actuation had occurred, added to the confusion of the situation and unnecessarily increased the burden on the operators. Operators diagnosed plant conditions and appropriately disregarded these indications. The safety injection annuciator will always incorrectly alarm on a loss of ac power. This is a design deficiency. The cause of the spurious indication on both safeguard load sequencer system panels is still unknown.

14. The operating staff, with the concurrence of management, did not follow appropriate procedures when troubleshooting the electrical ground. Their actions unnecessarily delayed entry into Technical Specification Action Statement requirements that could require plant shutdown.

15. Once the electrical ground was located on the feeder from auxiliary transformer C to bus 1C, the operators did not aggressively pursue isolating the auxiliary transformer. Instead, they opted to leave the transformer energized while technicians performed inspections that did not require the transformer to be energized.

16. The operators' actions, after the transformer trip, were consistent with their training. However, in the Team's judgment, some operators lacked detailed plant knowledge in the following areas:

- Cautions associated with paralleling transformers.
- Requirements for resetting unit generator trips.
- The process for operating 220KV circuit breakers.
- Expected indications and timing of the loss of voltage automatic transfer scheme.
- Setpoints for residual heat removal system pressure interlock.
- Expected indication and meaning of lights on SLSS sequencer panels.
- Operability of diesel generators with auxiliary transformer C reactor coil bypass breakers removed.

These deficiencies may be due to inadequate operator training and/or procedures.

17. On occasion, some site personnel who generally evaluate plant data lacked a sufficiently inquiring attitude. As a result, certain significant indications of underlying reasons for system response or component performance were not detected until brought to the attention of SCE by the Team. It appears that SCE's process for evaluating and following up events may not be sufficiently thorough and systematic to assure that failed components are detected and adequately explained.

18. The status of the steam generator blowdown system is not indicated in the control room. The reestablishment of blowdown when the radiation monitors were reset was not recognized and adversely contributed to the cooldown of the reactor coolant system and to the delay in recovering the steam generator levels.

19. During the loss of all inplant ac power, sufficient information was available in the control room to enable the operators to follow their procedures and ensure plant safety. However, control room operators had failed to have the Technical Support Center computer reset following electrical ground troubleshooting activities. This failure disabled the computer's ability to record new plant data and thereby denied the operators access to pre-trip and post-trip trends that would have assisted real time and post-event analysis and evaluation. Had the station blackout been of longer duration, or involved additional complications, operator responses and the functions provided by the Technical Support Center could have been hampered by the lack of trend data.

20. Station maintenance records are incomplete, difficult to locate and, when available, lack sufficient detail to determine what was done.

21. The spurious ringing of the NRC red phone at the beginning of the event has not been explained, but it distracted control room personnel and contributed to the confusion in the communications between SCE and NRC.

22. ENS communications between NRC and SCE were not effective because: (1) the NRC Duty Officer was not knowledgeable about the unique design of the plant and, therefore, misinterpreted operator responses to questions; (2) communications with the plant were initially limited because statements by plant operators incorrectly implied that sufficient personnel were not available to support the establishment of an open line; (3) NRC asked leading questions and operators sometimes did not correct, and in some cases appeared to confirm, inaccurate information; (4) NRC questions characteristically focused on details rather than on the "big picture"; (5) NRC cluttered the communications channel with repetitive discussions about the sequence of events as additional NRC personnel came on the line to the exclusion of obtaining new plant information; (6) NRC resident inspectors relieved more knowledgeable plant operators as ENS communicators and reestablished communications at a location remote from real time plant information; and, (7) plant operators failed to inform the NRC of the declaration of an Unusual Event.

23. There were two malfunctions of the automated security access control equipment; however, site personnel implemented appropriate planned compensatory measures, thereby precluding a safety-safeguards interface problem.

24. There was no significant release of radioactivity.

It must be recognized that this report was compiled prior to completion of all required inspections and evaluations of equipment involved in the event. SCE's continuing diagnostic efforts have unearthed additional information nearly daily; however, this information has been easily integrated into the Team's understanding of the incident and in most cases has confirmed long-held hypotheses on the sequence of events. Future reports from SCE will incorporate the findings of those studies which are not yet complete.

PRINCIPAL FINDINGS AND CONCLUSIONS

THE EVENT WAS SIGNIFICANT

- ALL INPLANT AC POWER WAS LOST FOR FOUR MINUTES
- ALL STEAM GENERATOR FEEDWATER WAS LOST FOR THREE MINUTES
- A SEVERE WATER HAMMER WAS EXPERIENCED IN THE FEEDWATER SYSTEM
  - CAUSED A LEAK
  - DAMAGED PLANT EQUIPMENT
  - CHALLENGED THE INTEGRITY OF THE ULTIMATE HEAT SINK
- ALL INDICATED STEAM GENERATOR WATER LEVELS DROPPED BELOW SCALE
- THE REACTOR COOLANT SYSTEM EXPERIENCED AN ACCEPTABLE BUT UNNECESSARY COOLDOWN TRANSIENT

T. T. MARTIN, 488 1280
JANUARY 22, 1986

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FEB 04 1986

MEMORANDUM FOR: Harold R. Denton, Director, NRR
James M. Taylor, Director, IE
John B. Martin, Regional Administrator, Region V

FROM: Victor Stello, Jr.
Acting Executive Director
of Operations

SUBJECT: STAFF ACTIONS RESULTING FROM THE INVESTIGATION OF THE NOVEMBER 21 SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 1 EVENT (NUREG-1190)

An advance copy of the subject report was transmitted to you by memorandum dated January 20, 1986 from the San Onofre Team Leader, Thomas T. Martin. The report documents the Team's efforts in identifying the circumstances and causes of the November 21, 1985 event, together with findings and conclusions which form the basis for identifying follow-on actions.

You will note from the report that the licensee has not completed troubleshooting and the determination of root causes for all equipment failures or malfunctions. Consequently, the results of future troubleshooting or analysis activities may form the basis for additional follow-on actions. The identification of these additional actions is a responsibility of the normal program office. The responsibility for the followup and reporting on the licensee's continued troubleshooting and determination of root cause for equipment failures is Region V.

The purpose of this memorandum is to identify and assign responsibility for generic and plant-specific actions resulting from the investigation of the San Onofre event (documented in NUREG-1190). In this regard, you are requested to review the enclosure which specifies staff actions resulting from the investigation of the San Onofre event. You are requested to determine the actions necessary to resolve each of the items in your area of responsibility and, where appropriate, identify additional staff actions or revisions as our review and understanding of this event are refined. Plant-specific actions required for plant restart should receive priority attention.

In view of the importance of this subject, I intend to closely monitor the resolution of these items. By March 1, 1986, please provide a written summary of the schedule and status of each item within your responsibility listed in the enclosure or that you have identified. Further, I request that you prepare a written status report on the disposition of your items (and anticipated actions for uncompleted items) within three to six months. Every effort should be made to dispose of these items promptly.

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The enclosure is based on the Team's report and its presentation to the Commission on January 22, 1986. Accordingly, it does not include all licensee actions, nor does it cover NRC staff activities associated with normal event followup such as authorization for restart, plant inspections, or possible enforcement items. These items are expected to be defined and implemented in a routine manner. Overall lead responsibility for staff actions relating to facility restart is separate from this effort and rests with Region V. Additionally, RV is responsible for coordinating and promptly communicating the staff's requirements which must be resolved before operations at San Onofre may be resumed.

Victor Stello, Jr.
Acting Executive Director
for Operations

Enclosure:
As stated

cc w/enclosure:
J. Davis, NMSS
T. Murley, RI
J. N. Grace, RII
J. Keppler, RIII
R. Martin, RIV

STAFF ACTIONS RESULTING FROM THE INVESTIGATION OF THE NOVEMBER 21 SONGS-1 EVENT

(Reference: NUREG-1190)

1. Item: Adequacy of feedwater check valves to perform safety function. (References: Commission briefing, Sections 6.2.4, 6.4, 6.7, and Principal Finding)

Action	Responsible Office	Category
Implement and coordinate the staff and industry actions necessary to assure the reliability of safety-related check valves. Other offices to assist as requested. Areas to be evaluated include: - licensee's engineering report on root cause analysis and proposed corrective actions - adequacy of check valve design for this application - adequacy of Inservice Testing (IST) Program and procedures to detect degraded and failed valves - adequacy of check valves (and related testing programs) in other systems such as RHR system	IE	Plant-specific Generic

2. Item: Completeness of resolved USI A-1, "Water Hammer". (References: Finding numbers 1, 2, 3, 8 and 9)

Action	Responsible Office	Category
Assess the need to re-evaluate USI A-1 to specifically address the potential for and prevention of condensation-induced water hammers in feedwater piping (assume the issue concerning check valve integrity will be resolved in item 1).	NRR	Generic

3. Item: Adequacy of San Onofre Unit 1 design.
(Commission briefing, Finding numbers 11 and 13)

Action	Responsible Office	Category
Implement and coordinate the staff's actions to re-evaluate the following San Onofre design features: - manual loading of the diesel generators following a loss of power event - manual actuation of steam line isolation valves and assurance of steam generator availability to remove decay heat - lack of steam generator blowdown status in control room - adequacy of the licensee's design change to eliminate spurious SI indication on loss of power	NRR	Plant-specific

4. Item: Adequacy of post-trip review.
(References: Sections 6.6 and 7.2.2.4 and Finding number 17)

Action	Responsible Office	Category
a. Evaluate NRC requirements for ensuring that sufficient event data are retrievable to accurately reconstruct the event following a loss of offsite power.	NRR	Generic
b. Evaluate the licensee's process for post-trip review and evaluation, including the thoroughness of review and oversight provided by the onsite and offsite nuclear safety review groups.	Region V	Plant-specific

5. Item: Adequacy of licensee's recordkeeping practices. (References: Section 6.5 and Finding number 20)

Action	Responsible Office	Category
Evaluate the adequacy of the licensee's maintenance records.	Region V	Plant-specific

6. Item: Adequacy of operator training and/or procedures. (References: Section 7 and Finding numbers 14, 15 and 16)

Action	Responsible Office	Category
Review the implementation of the training program regarding operator understanding and actions in the area of electrical systems, and invoking technical specification action statements.	Region V	Plant-specific

7. Item: Adequacy of emergency notifications and NRC response. (References: Section 7.3 and Finding number 22)

Action	Responsible Office	Category
a. Verify the adequacy of the licensee's procedures and training for reporting of events to NRC Operations Center.	Region V	Plant-specific
b. Evaluate the need for changes in NRC policy or guidance regarding: the use of the ENS line; the use of NRC personnel as ENS communicators; and possible approaches to improve the ability to determine the overall plant status.	IE	Generic

8. Item: Significance of backlog of license amendments. (Reference: Commission briefing)

Action	Responsible Office	Category
Evaluate whether a backlog of license amendments and technical specification changes contributed to delays in approving the licensee's IST program.	NRR	Plant-specific

ATTC

EQUIPMENT PROBLEMS

ITEM		NATURE OF FAILURE	PROBABLE ROOT CAUSE	COMMENTS
1.	POWER SUPPLY CABLE	GROUND FAULT	WATER INLEAKAGE	ANALYSES ONGOING
2.	FLASH EVAPORATOR UNIT	TUBE/SHELL RUPTURED	OVERPRESSURIZED DUE TO FAILED CHECK VALVE	NO LONGER USED
3.	SAFETY INJECTION ANNUNCIATOR	SPURIOUS ALARM	LOSS OF POWER	DESIGN INADEQUACY
4.	SAFEGUARD LOAD SEQUENCING SYSTEM	INDICATED SAFETY INJECTION ACTUATION	NOT IDENTIFIED	ANALYSES ONGOING
5.	LOSS OF VOLTAGE AUTO TRANSFER SCHEME	FAILED TO REALIGN CIRCUIT BREAKERS TO RESTORE POWER	NOT IDENTIFIED	ANALYSES ONGOING

T. T. MARTIN, 488 1280
JANUARY 22, 1986

EQUIPMENT PROBLEMS CONT'D

ITEM		NATURE OF FAILURE	PROBABLE ROOT CAUSE	COMMENTS
6.	FOX III COMPUTER	NO RECORDED DATA BEFORE/AFTER TRIP	POWER INTERRUPTION	RESET REQUIRED
7.	TURBINE RUPTURE DISKS (4 OF 8)	RUPTURED	OVERPRESSURIZED DUE TO LOSS OF POWER	EXPECTED
8.	EMERGENCY NOTIFICATION SYSTEM	SPURIOUS RINGS	NOT IDENTIFIED	CANNOT REPRODUCE
9.	RCP THRUST BEARING	HIGH TEMPERATURE ALARM	FAILED DETECTOR	
10.	CHECK VALVE FWS-378	STUDS STRETCHED BODY TO BONNET LEAK	WATER HAMMER	

T. T. MARTIN, 488 1280
JANUARY 22, 1986

EQUIPMENT PROBLEMS CONT'D

ITEM	NATURE OF FAILURE	PROBABLE ROOT CAUSE	COMMENTS
11. CHECK VALVE FWS-345	NUT MISSING DISC SEPARATED FROM HINGE ARM	NOT IDENTIFIED	EVALUATION ONGOING
12. CHECK VALVE FWS-346	NUT MISSING DISC SEPARATED FROM HINGE ARM	NOT IDENTIFIED	EVALUATION ONGOING
13. CHECK VALVE FWS-398	NUT LOOSE STUCK-OPEN	NOT IDENTIFIED	EVALUATION ONGOING
14. CHECK VALVE FWS-438	NUT LOOSE STUCK-OPEN	NOT IDENTIFIED	NOT PINNED EVALUATION ONGOING

T. T. MARTIN, 488 1280
JANUARY 22, 1986

EQUIPMENT PROBLEMS CONT'D

ITEM	NATURE OF FAILURE	PROBABLE ROOT CAUSE	COMMENTS
15. CHECK VALVE FWS-439	NUT LOOSE STUCK-OPEN	NOT IDENTIFIED	NOT PINNED EVALUATION ONGOING
16. FLOW CONTROL VALVE FCV-457	BROKEN YOKE BENT STEM	WATER HAMMER	INERTIA
17. B STEAM GENERATOR FEEDWATER LINE	CRACKED BENT DENTED	WATER HAMMER	EVALUATION ONGOING BEING REMOVED MOST SUSCEPTIBLE PIPING
18. FEEDWATER LINE SUPPORTS & SNUBBERS	DAMAGED	WATER HAMMER	EVALUATION ONGOING

T. T. MARTIN, 488 1280
JANUARY 22, 1986

EQUIPMENT PROBLEMS CONT'D

ITEM	NATURE OF FAILURE	PROBABLE ROOT CAUSE	COMMENTS
19. AUXILIARY FEEDWATER LINE SUPPORTS	DISPLACEMENT	WATER HAMMER	
20. CONTAINMENT SPHERE	SMALL CRACK-LIKE INDICATIONS	NOT IDENTIFIED	EVALUATION ONGOING
21. SECURITY ACCESS CONTROLS	SECURE	SAFEGUARDS INFORMATION	NO SAFETY/SAFE-GUARDS INTERFACE PROBLEM

T. T. MARTIN, 488 1280
JANUARY 22, 1986

AD

MODIFICATIONS ADDRESSING FEEDWATER WATER HAMMER

- o REPLACE 8 PACIFIC CHECK VALVES WITH ATWOOD-MORRILL VALVE DESIGN

- o PROVIDE IMPROVED IN-SERVICE TEST CAPABILITY FOR REPLACEMENT VALVES

- o SLOPE "B" FEEDWATER PIPE INSIDE CONTAINMENT (always from SG)

- o ADD CHECK VALVES, WITH IST PROVISIONS, IN EACH FEEDWATER LINE INSIDE CONTAINMENT

- o PROVIDE FOR AUTOMATIC CLOSURE OF MAIN FEEDWATER FCV'S FOLLOWING UNIT TRIP AND LOSS OF MFW PUMPS

- o EVALUATE ADDITION OF VOID SENSING DEVICE WITH CONTROL ROOM ALARM

GENERAL MODIFICATIONS

- INSTALL CONTROL ROOM INDICATION OF BLOWDOWN STATUS*
- PROVIDE AUTOMATIC BLOWDOWN ISOLATION ON AFW ACTUATION*
- EVALUATE POSSIBILITY OF PROVIDING FOR AUTOMATIC LOADING OF DIESEL GENERATORS ON LOSS OF POWER**
- EVALUATE ALTERNATIVES FOR PROVIDING AN IMMEDIATELY AVAILABLE BACKUP SOURCE OF OFFSITE POWER**
- INSTALL ADDITIONAL PROTECTION FOR AUXILIARY TRANSFORMER FEEDER CABLING*
- MODIFICATIONS TO PRECLUDE SPURIOUS ANNUNCIATION OF SAFETY INJECTION*
- MODIFY TSC COMPUTER FOR AUTOMATIC RESTART FOLLOWING POWER RESTORATION*
- EVALUATE PROVISION OF UPS FOR TSC COMPUTER**
- EVALUATE PROVISION OF UPS FOR BUSSES SUCH AS VITAL BUS NO. 4**

GENERAL MODIFICATIONS (CONTINUED)

- DETERMINE AND CORRECT CAUSE OF SPURIOUS RED PHONE RINGING**
- INSTALL CONTROL ROOM CLOCK NOT DEPENDENT ON AC POWER*
- REVIEW CONTROL ROOM INDICATION OF VITAL BUS STATUS (CRDR)*
- MODIFY CURRENT LIMITING BYPASS REACTOR BREAKER DESIGN TO PERMIT DG TO CLOSE ONTO 4 KV BUS WHEN BREAKER IS RACKED OUT*

NOTES: *TENTATIVELY PLANNED FOR CURRENT OUTAGE

**SCHEDULE NOT YET ESTABLISHED