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AUTH.NAME AUTHOR AFFILIATION
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SUBJECT: LER 88-034-00:on 881215,safety related CCWS valves
susceptible to seismically induced common mode failures.
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LICENSEE EVENT REPORT (LER)

Facility Name (1)	Docket Number (2)	Page (3)
SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2	0 5 0 0 0 3 6 1	1 of 0 9

Title (4)

SAFETY RELATED COMPONENT COOLING WATER SYSTEM VALVES SUSCEPTIBLE TO SEISMICALLY-INDUCED COMMON MODE FAILURES

EVENT DATE (5)			LER NUMBER (6)					REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
Month	Day	Year	Year	///	Sequential Number	///	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
1 2	1 5	8 8	8 8	---	0 3 4	---	0 0	0 1	1 6	8 9	SONGS, UNIT 3	0 5 0 0 0 3 6 2
												0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

	20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
	20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
	20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		Other (Specify in Abstract below and in text)
	20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		
	20.405(a)(1)(iv)	X	50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
	20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER	
	AREA CODE	
H. E. Morgan, Station Manager	7 1 4	3 6 8 - 6 2 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

		Expected Submission Date (15)	Month	Day	Year
Yes (If yes, complete EXPECTED SUBMISSION DATE)	XX NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On 12/15/88, with Units 2 and 3 operating at 100% power, as a result of a recent evaluation of the Component Cooling Water System (CCWS), it was determined that the CCWS did not meet its design basis. Specifically, 15 safety related valves in the CCWS are provided with non-1E control circuits which might spuriously operate in conjunction with a Design Basis Earthquake (DBE), creating a common mode failure that could prevent the fulfillment of the system function. This condition is contrary to 10 CFR 50, Appendix A, Criterion 2 and 44. When this condition was identified, it was determined that plant shutdown was not warranted because of the low probability of an earthquake in combination with the low probability of the postulated spurious operations of the control devices in question. Results from the subsequent seismic testing or analysis of the various devices and cabinets involved form the basis for the justification for continued operation.

These valves are safety related for a pressure boundary function only, and are not required to operate during a design basis event. As part of the original system design, these components were previously assumed to lose electrical power during a seismic event and fail in the safe position, resulting in continued system function. Consequently, the design of these valves did not require their control devices to be seismically qualified. The root cause is related to deficiencies with programs for establishing and controlling design basis documentation.

The CCWS as well as other safety systems will be subjected to reviews for similar concerns. Corrective actions being taken for the root cause concerns, are addressed in a 10/3/88 submittal to the NRC regarding SCE's assessment of engineering and technical support for San Onofre.

Plant: San Onofre Nuclear Generating Station
Units: Two and Three
Reactor Vendor: Combustion Engineering
Event Date: 12-15-88

A. CONDITIONS AT TIME OF THE EVENT:

Unit 2: Mode 1 (100% reactor power operation)
Unit 3: Mode 1 (100% reactor power operation)

B. BACKGROUND INFORMATION:

The Component Cooling Water System (CCWS) (EIIS System Code CC) has two redundant trains (critical loops) that supply cooling water to redundant trains of safety equipment needed for plant shutdown and emergency cooldown subsequent to a design basis event. A non-critical loop (NCL), which is aligned to either one of the two critical loops during normal operation, supplies cooling water to equipment and components for normal plant operation and for normal plant shutdown.

Consistent with 10 CFR 50, Appendix A, General Design Criteria (GDC) 2 and 44, the design basis of the CCWS includes the requirement to be capable of performing its safety function following a Design Basis Earthquake (DBE) concurrent with a single active failure.

Each Component Cooling Water (CCW) train contains a surge tank (EIIS Component Code TK) which has low level and low-low level setpoints. At the low level setpoint, the associated surge tank makeup supply valve from the Nuclear Service Water (NSW) System (non-safety related and non-seismically qualified) (EIIS System Code KG) automatically opens. At the low-low level setpoint, the NCL isolation valves (EIIS Component Code ISV) automatically close to terminate the loss of CCWS water inventory due to an assumed fault on the NCL following a DBE or High Energy Line Break Accident (HELBA).

With regard to CCWS water inventory requirements, the CCWS must be provided with (or have available) a Seismic Category I makeup water source or be capable of operating for 7 days without makeup following a design basis event. As indicated in response to FSAR question 010.49, the CCWS design leakage was analyzed to be sufficiently small to allow operation for 122 days without makeup; therefore, no seismically qualified makeup capability was provided. To meet this design criteria, critical loop leakage must be less than or equal to .142 gpm. In 1983, it was realized that the CCWS leakage criteria (.142 gpm) was overly restrictive such that 1) the system could not have been practically maintained to ensure the criteria was met, and 2) minor leakage exceeding the criteria could not have been practically detected. In 1984, a design modification was implemented and appropriate operating instructions were revised which provided the capability to supply makeup water to the CCW surge tank from existing seismically qualified mobile fire tankers (EIIS Component Code TK)(EIIS System Code KP) (primary function is for fire suppression water supply).

Between May 2 and June 10, 1988, a Safety System Functional Inspection (SSFI) was conducted by the NRC. The SSFI assessed the operational readiness of the Units 2 and 3 CCW and Salt Water Cooling (SWC) systems under normal and analyzed accident conditions. One of the SSFI findings involved the failure to perform an analysis of the adverse effects of earthquakes on the design of the valve motor operator control circuits for the CCW surge tank outlet block valve (EIIS Component Code SHV). The CCW surge tank motor operated outlet block valves are not supplied by a Class 1E power supply. Contrary to GDC 2 and 44, there was a remote possibility that the valves would close spuriously during a DBE, creating a common mode failure that could prevent the fulfillment of the system function. This SSFI finding, among others, was later the subject of a Notice of Violation, included with the NRC's inspection report dated August 3, 1988.

On June 17, 1988, the overload devices were pulled from the CCW surge tank motor operated isolation valve feeder breakers. This action precludes the possibility of a common mode failure of the breaker relays, which could cause inadvertent closure of the surge tank isolation valves during an earthquake.

At the end of the SSFI, because a number of findings involving CCWS design bases discrepancies were identified, SCE committed to perform an operability evaluation of the CCWS. Results of a preliminary system review, indicating that the CCWS is operable, was submitted to the NRC in a letter dated June 24, 1988. In this letter, SCE committed to perform a more comprehensive system review, including a reassessment of the potential failure modes and effects of each major component and instrument/electrical subsystem. SCE's response to the NOV in a letter dated September 2, 1988, reiterated this commitment. Results of this effort were submitted in a report to the NRC dated December 16, 1988.

C. DESCRIPTION OF THE EVENT:

1. Event:

On December 15, 1988, with Units 2 and 3 operating at 100% power, as a result of the recent evaluation of the CCWS, it was determined that the CCWS did not meet its design basis. Specifically, safety related systems should be designed to withstand the effects of natural phenomena such as earthquakes. Contrary to this, 15 valves in the CCWS are provided with non-1E control circuits which might spuriously operate in conjunction with a DBE.

The CCWS functions to remove heat from both safety and non-safety related components. Components for the critical loop portion of the CCWS are served by a combination of 1E and non-1E power supplies. The essential components, those which must change position or operate post-accident, are powered from 1E sources. Some components which are safety related for pressure boundary only and are not required to operate are provided with non-1E, non-seismic power. As part of the original system design, these components were assumed to lose electrical power during a seismic event and fail in the safe position, resulting in continued system function. During the Failure Modes and Effects Analysis (FMEA) portion of the CCWS review, it was determined that a more conservative methodology than that used in the original FMEA may be appropriate which could result in a design basis deficiency.

Specifically, it is postulated that non-1E power may not be lost after an earthquake and furthermore the safety related valves with non-1E control circuits have the potential to spuriously operate to a position other than that desired. Under this scenario, the CCWS may not be capable of performing its function in accordance with the revised methodology.

The FMEA review effort identified twenty-three safety related valves having non-1E power and control circuits whose failure could cause spurious valve actuation resulting in loss of CCW critical loop function or cross-train leakage. In order to satisfy 10 CFR 50, Appendix R requirements, eight of these valves are provided with power lockout or power removal, preventing spurious actuation during a DBE. The remaining 15 valves in question are listed below.

Valve Description	Valve Nos.	Valve Operator Type
CCW Pump Mini-flow Isolation Valves	HV-6220, 6551 HV-6221, 6552	Motor
CCW Supply and Return Valves for Letdown HX	HV-6293A, 6293B HV-6522A, 6522B	Air
CCW Mini-flow Control Valves	HCV-6537, 6539 HCV-6538	Air
CCW Swing Pump (P-025) Suction and Discharge Valves	HV-6222B, 6224B HV-6226B, 6228B	Motor

When this condition was identified on December 15, 1988, it was determined that a significant safety concern did not exist and that plant shutdown was not warranted because of the low probability of an earthquake in combination with the low probability of the postulated spurious operations of the control devices in question. To confirm this determination, an extensive operability assessment of this condition was immediately undertaken. This effort, which included seismic testing or analysis of the various devices and cabinets involved, forms the basis for the justification for continued operation (JCO). The JCO was completed on December 22, 1988. A summary of this JCO is provided under Section F below.

As discussed above (Section B), a design modification was implemented in 1984 which provided the capability to supply makeup water to the CCW surge tank from existing seismically qualified mobile fire tankers. In November, 1988, during the recent operability evaluation of the CCWS, it was determined that due to the potential for a high radiation field several hours after a postulated Loss of Coolant Accident (LOCA) in the vicinity of the CCW Surge Tank, the actions necessary to connect the makeup hoses to the CCWS could be hindered. As an enhancement to CCW makeup capability, the fire truck connection points on the CCWS were relocated upstream of the air-operated Letdown Heat Exchangers (LDHX) inlet isolation valves where radiation levels are postulated to be low, and appropriate operating

procedures were revised. On December 23, 1988, while evaluating possible compensatory measures in response to the identification of the potential for seismically-induced common mode failures, it was determined that the CCWS makeup connections upstream of the LDHX inlet isolation valves are not appropriate since these valves, which are powered by a non-1E electrical source, are postulated to close following a DBE. The CCWS was judged to be operable, however, since makeup to the CCWS is required to be initiated within 4 to 6 hours under the postulated accident scenario, and appropriate LDHX isolation valves could have been easily opened by utilizing nitrogen bottles which are commonly used for calibration of the air-operated valves. Alternative locations were established on the CCWS for the connection of the fire trucks which would not be affected in the event of a DBE.

2. Inoperable Structures, Systems or Components that Contributed to the Event:

   None.

3. Sequence of Events:

   Not applicable.

4. Method of Discovery:

   This condition was discovered as a result of performing an FMEA review of CCWS components in response to a commitment made to the NRC.

5. Personnel Actions and Analysis of Actions:

   Not applicable.

6. Safety System Responses:

   Not applicable.

D. CAUSE OF THE EVENT:

1. Immediate Cause:

   The design of the subject CCW valves' control circuits did not: 1) include analyses of adverse effects of postulated earthquakes; and 2) reflect the combination of the effects of these valves spuriously operating in conjunction with a safe shutdown earthquake. This potential for seismically-induced common mode failures is contrary to 10 CFR 50, Appendix A, Criterion 2 and 44.

2. Intermediate Cause:

   LER 88-008, Revision 1 (Docket No. 50-361), reported a similar condition involving the CCW surge tank isolation valves. The automatic closure of the CCW surge tanks motor operated valves (MOV) on low-low surge tank level is not required as a safety function. Because this function was not considered a safety function, the associated power and control circuits were not designed to Class 1E requirements. Additionally, the remote possibility of a common mode failure (earthquake) of control relays in the motor control center (MCC) resulting in valve closure was not identified when the system was designed. This was primarily because the methodology employed at that time assumed that non-1E power would be lost during an SSE, thereby precluding spurious operation. Consequently, no evaluation of the occurrence was performed and the relays were not required to be seismically qualified. For this same reason, the control devices for the 15 valves being reported in this LER were not seismically qualified.

3. Root Cause:

   The root causes of the conditions described in this LER are related to deficiencies with establishing and controlling design basis documentation. These root causes can be summarized as follows;

   a. The design bases for the CCW System were not defined in sufficient detail to identify the logic that the associated analyses were based upon.

   b. The programmatic requirements for identification and control of the design basis is not clear. Such requirements are contained in numerous procedures making implementation difficult.

   c. The organizational responsibility for system design basis, licensability, functionability and operability is divided among multiple organizations causing difficulty in controlling the design process.

   In response to general NRC concerns in this area as a result of the SSFI, SCE conducted a thorough review of it's programs for controlling design, engineering and technical related matters as they pertain to San Onofre. The principle conclusions and recommendations of this study are identified in SCE's October 3, 1988 letter to the NRC regarding this matter. This study concluded that the problems reported herein resulted from (1) the complexity of the current organization, (2) heavy reliance on engineering contractors combined with inadequate allocation of SCE engineering resources, and (3) the lack of readily accessible comprehensive design basis documentation.

E. CORRECTIVE ACTIONS:

1. Corrective Actions Taken:

a. An engineering review, including seismic testing or engineering analysis, was performed for each of the non-seismically qualified devices in question. The results of this review provide the basis for a JCO.

b. Until the above review was completed, operators were alerted (via a Special Order issued on 12/16/88) to the possibility of spurious valve operation and the actions necessary to mitigate the consequences of these spurious actuations.

2. Planned Corrective Actions:

a. As described in the above root cause discussion, the principle conclusions and recommendations are identified in SCE's October 3, 1988 submittal to the NRC. Corrective actions to address these conclusions include (1) a re-organization with responsibility for design functions and the design basis focused in one department, (2) augmentation of in-house engineering resources and performance of the majority of conceptual engineering in-house, and (3) the establishment of a design basis documentation (DBD) program to recapture and maintain the design basis.

b. The review of CCWS non-1E devices performed thus far is based, in part, on a demonstration of similarity between many of the devices installed in the affected non-seismically qualified circuits and devices installed in qualified circuits. SCE will thoroughly review the application of similarity for each device using the methodology established in NRC Generic Letter 87-02, Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue A-46. The scope of this review is to verify compliance with GDCs 2 and 44 as they apply to the CCWS. This review is expected to be completed by July 1, 1989.

c. Each safety related system will be reviewed for similar concerns in conjunction with the preparation of design bases documents under the DBD Program identified in action 2.a above. The DBD program will integrate the plant design criteria with existing design and as-built information. The DBD program is a 5 to 6 year program described in our letter of January 9, 1989. This schedule is considered acceptable given the favorable results of our review of non-qualified devices on CCWS safety related valves.

F. SAFETY SIGNIFICANCE OF THE EVENT:

Since the 15 safety related CCWS valves contain circuit devices which are not seismically qualified, there was a possibility that these valves could spuriously operate during a DBE. A scenario involving spurious opening or closing of one or more of these valves would result in the inability of the CCWS to perform its safety function. However, based on the seismic testing and/or engineering review of the devices and cabinets involved, there is a high degree of confidence that none of the postulated scenarios would occur during a DBE. This conclusion is supported by the JCO summarized below.

JCO Summary:

For each of the subject valves, the devices comprising the control and power circuits were compiled from a review of design documents and drawings. For each device which could affect the ability of the CCWS to perform its safety function, the manufacturer and model number were determined. If the model numbers of the devices were identical to those procured for seismically qualified 1E circuits, then spurious actuation was not considered credible. Each device was subjected to additional review if its model number was not identical to that of a qualified device (listed below). Verification by walkdown was performed to confirm these components were actually in use in the affected circuits.

Cutler Hammer Motor Starters, Type A-1, Size 1 NEMA, Model No. C50CN3
General Electric Relays - CR120 Series
Namco Limit Switches - EA170 Series
Cutler Hammer Control Switch E30AB

To verify the hypothesis of spurious actuation due to relay chatter in non-seismically qualified electrical components during a DBE, seismic testing of specific components was undertaken. These components included three of the Cutler Hammer (CH) starter assemblies and three of the General Electric (GE) relays. Guidelines of ANSI/IEEE C37.98, 1984, were used to perform the seismic testing. Additional test runs were made using site/building specific Updated Final Safety Analysis Report response spectra, with amplifications of 6 to 8 in the 5 to 10 Hz frequency range to account for natural frequencies in the MCCs and relay cabinets. As part of the test, the auxiliary contacts of the starter were connected to the GE relay to replicate wiring in the control circuit. Main contactor chatter and seal-in, in addition to chatter of the GE relay, were monitored in the shake test. From this testing, it was concluded that the failure mode hypothesized in the FMEA is not credible.

A review of the mounting details for the MCC cabinets housing the CH motor starters and the auxiliary relay panels housing the GE relays was performed. This review revealed that the cabinets and panels will maintain their structural integrity during a DBE.

Analyses of the limit switches and the control switch were performed to determine whether their moveable components could change position during a DBE. This analysis shows that these switches can withstand an acceleration of about 5 to 6 times greater than that assumed for a DBE. Therefore, failure of these components due to a DBE is not considered credible.

G. ADDITIONAL INFORMATION:

1. Component Failure Information:

   Not applicable.

2. Previous LERs on Similar Events:

   Recent LERs reporting similar design and design control related conditions:

   <u>Unit 1 (Docket No. 50-206)</u>

   LER 1-87-015 reported that certain systems were susceptible to single failure.

   LER 1-88-009 reported a condition in which the emergency diesel generators could have exceeded an intended electrical load limit.

   LER 1-88-006 reported a condition where the Unit 1 Backup Nitrogen Systems (as designed, installed and operated) did not satisfy the licensing and design basis for the systems.

   LER 1-88-001 reported that several components requiring environmental qualification were not included in the administrative controls for the environmentally qualified equipment. Additionally, other components were found to be in an unqualified configuration.

   <u>Unit 2 (Docket No. 50-361)</u>

   LER 2-88-017 reported that a spent fuel pool siphon event occurred as a result of the failure to identify and implement the design intent to utilize administrative controls on certain locked valves.

   LER 2-88-010 reported a condition in which both emergency chillers were rendered inoperable as a result of not addressing freon level as a critical design parameter.

   LER 2-88-008 reported various conditions resulting in the Component Cooling Water System being outside its design basis due to design control program deficiencies.

3. Results of NPRDS Search:

   Not applicable.

Southern California Edison Company

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January 16, 1989

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Subject: Docket No. 50-361
30-Day Report
Licensee Event Report No. 88-034
San Onofre Nuclear Generating Station, Unit 2

Pursuant to 10 CFR 50.73(a)(2)(ii), this submittal provides the required 30-day written Licensee Event Report (LER) for an occurrence involving the Component Cooling Water (CCW) Systems in Units 2 and 3. Since this occurrence involves similar systems, cause, and corrective actions applicable to Units 2 and 3, a single report for Unit 2 is being submitted in accordance with NUREG-1022. Neither the health and safety of plant personnel nor the health and safety of the public was affected by this occurrence.

If you require any additional information, please so advise.

Sincerely,

H E Morgan

Enclosure: LER No. 88-034

cc: F. R. Huey (USNRC Senior Resident Inspector, Units 1, 2 and 3)
J. B. Martin (Regional Administrator, USNRC Region V)
Institute of Nuclear Power Operations (INPO)

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