Southern California Edison Company

SAN ONOFRE NUCLEAR GENERATING STATION

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January 26, 1990

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

Subject: Docket No. 50-206
30-Day Report
Licensee Event Report No. 89-030
San Onofre Nuclear Generating Station, Unit 1

Pursuant to 10 CFR 50.73(d), this submittal provides the required 30-day written Licensee Event Report (LER) for an occurrence involving the potential diversion of Emergency Core Cooling System Recirculation Flow to the Reactor Coolant System. Neither the health and safety of plant personnel or the public was affected by this occurrence.

If you require any additional information, please so advise.

Sincerely,

H E Morgan

Enclosure: LER No. 89-030

cc: C. W. Caldwell (USNRC Senior Resident Inspector, Units 1, 2 and 3)

J. B. Martin (Regional Administrator, USNRC Region V)

Institute of Nuclear Power Operations (INPO)

IE22

LICENSEE EVENT REPORT (LER)

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

Title (4)

POTENTIAL DIVERSION OF EMERGENCY CORE COOLING SYSTEM RECIRCULATION FLOW THROUGH A SPRING-LOADED CHECK VALVE

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	2 8	8 9	8 9	--- 0 3 0	--- 0 0		0 1	2 6	9 0	NONE	0 5 0 0 0
											0 5 0 0 0

OPERATING MODE (9)	1
POWER LEVEL (10)	0 9 2

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)	X 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
H. E. Morgan, Station Manager	AREA CODE 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 December 28, 1989, at 1040, it was determined that previous evaluations of post-Loss of Cooling Accident Emergency Core Cooling System (ECCS) recirculation flow from the containment sump to the reactor core incorrectly credited continued post-accident pressurization of the non-seismically qualified Volume Control Tank (VCT) to prevent a flow diversion of recirculation fluid to the VCT. This flow diversion occurs from the suction of the charging pumps to the VCT through a piping line, which contains a spring-loaded check valve (VCC-316) that is set at 75 psid. At one time, VCC-316 provided a thermal relief path for a positive displacement test pump located in parallel with the charging pumps. With the VCT assumed to be depressurized after the design basis seismic event and either one or both recirculation pumps operating, a continuous flow diversion of the recirculated water to the VCT would occur. This flow diversion could result in inadequate ECCS recirculation flow to the reactor core and, if not discovered and corrected, could lead to a condition in which containment sump fluid inventory was insufficient to maintain continued recirculation flow to the reactor core.

To preclude any flow diversion to the VCT through this line, an isolation valve downstream of VCC-316 was closed and is being administratively controlled in the closed position. An engineering evaluation will be performed to determine if the spring loaded check valve will be removed or modified.

The failure to recognize this diversion flow path is attributed to weaknesses in SCE's engineering and technical support for San Onofre, which is described in detail in our October 3, 1988, submittal to the NRC addressing this subject. The corrective actions identified in that submittal are also applicable to the causes of this event.

Plant: San Onofre Nuclear Generating Station
Unit: One
Reactor Vendor: Westinghouse
Event Date: 12-28-89
Time: 1040

A. CONDITIONS AT TIME OF THE EVENT:

Mode: 1, Power Operation at 92% Power

B. BACKGROUND INFORMATION:

Safety Injection System (See Figure 1):

The Safety Injection System (SIS) [BQ] is designed to mitigate core damage resulting from overheating following a loss of coolant accident (LOCA). Initially after a LOCA, the SIS injects borated water from the Refueling Water Storage Tank (RWST) [TK] through the SIS header and into the three Reactor Coolant System (RCS) [AB] loops (Loop A, Loop B, and Loop C) via separate SIS lines.

The Recirculation System (RS) [BP] provides core cooling for extended periods following design basis accidents and is initiated after a sufficient quantity of water has accumulated in the containment sump [WK]. Two recirculation pumps [P] take a suction from the containment sump and discharge water to a common header through the Recirculation Heat Exchanger [HX] to the suction of the two charging pumps [CB,P] for recirculation to the RCS and to the suction of the refueling water pumps [P] for recirculation to the containment spray system [BE]. The Chemical and Volume Control System (CVCS) positive displacement Test Pump (TP) [P] is installed in parallel with the two charging pumps and its suction piping is pressurized during the recirculation phase of Safety Injection (SI). This TP is used as a pressure source for hydrostatic testing of the RCS. Also, if both charging pumps are lost or removed from service, the TP can be used for RCS boric acid concentration control and to maintain normal seal water supply for the Reactor Coolant Pumps [P].

Chemical and Volume Control System (See Figure 1):

The Chemical and Volume Control System (CVCS) [CB] is designed to: 1) charge to and letdown from the RCS for chemical and volume control during normal operation; and 2) provide a means to assist in the recirculation of borated water in the RCS after a LOCA.

Two motor-operated isolation valves [ISV] located in the charging pump suction line from the RWST are provided to ensure suction is available to the charging pumps from the RWST in the event of a low Volume Control Tank

(VCT) [TK] level or a Safety Injection signal. When either of these valves open, an interlock will close the motor-operated VCT outlet isolation valve.

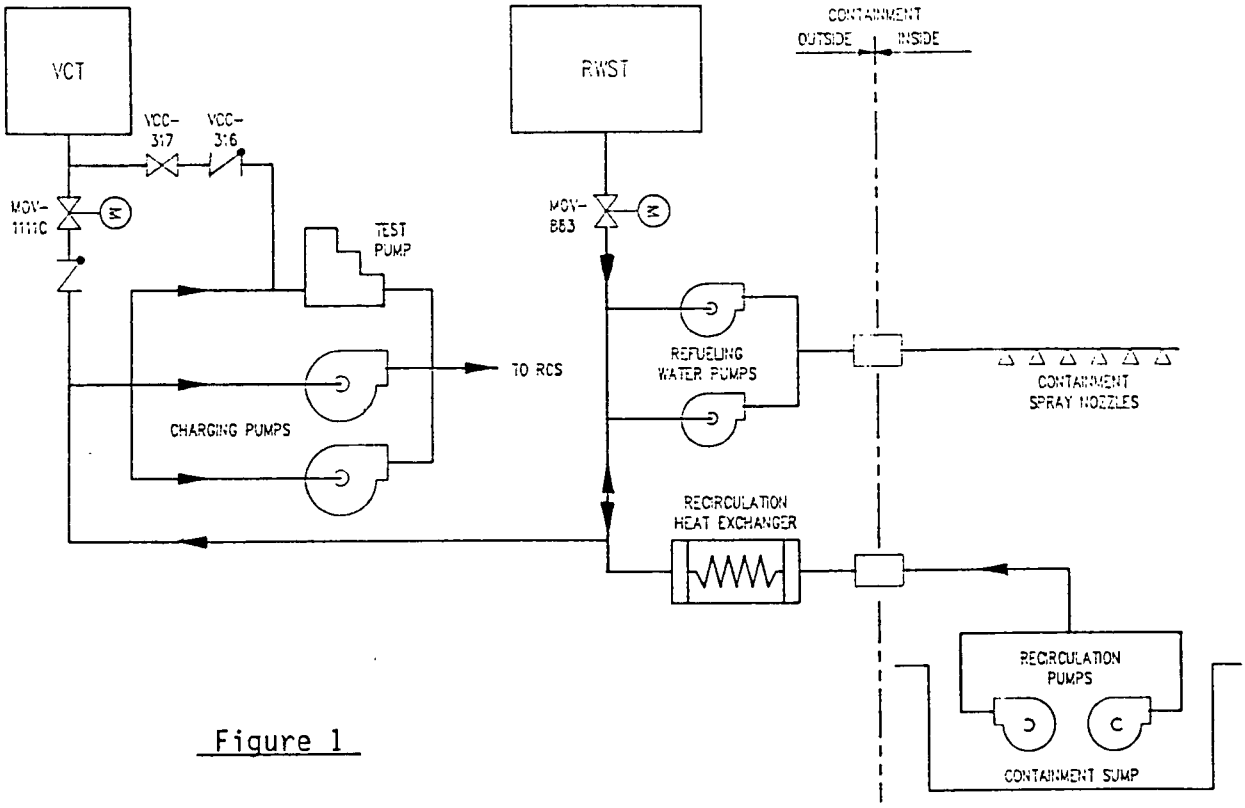

Figure 1

Spring-Loaded Check Valve (VCC-316) (See Figure 1):

A 1/2 inch line connecting the Test Pump (TP) suction line to the VCT upstream of the VCT outlet valve was installed during original plant construction to function as a thermal relief for the test pump suction piping if the test pump was operated with its discharge valve shut. The TP discharge relief valve originally relieved to the TP suction line. In 1983, the discharge of the TP discharge relief valve was rerouted to the Seal Water Heat Exchanger (SWHE) [HX] inlet line, which is cooled by Component Cooling Water (CCW) [CC]. Therefore, the thermal relief line is no longer necessary.

A spring loaded check valve (VCC-316) on the thermal relief line is designed to preclude the diversion of water from the test pump suction line to the VCT during the recirculation phase of SI. The check valve is set to open at a differential pressure of 75 pounds per square inch (psid) across the valve, which is greater than the calculated differential pressure between the suction line and the VCT.

C. DESCRIPTION OF THE EVENT:

1. Event:

On December 28, 1989, at 1040, it was determined that previous evaluations of post-LOCA recirculation flow from the containment sump to the reactor core incorrectly credited continued post-accident pressurization of the non-seismically qualified VCT to prevent diversion of recirculation fluid through the 1/2 inch TP thermal relief line. The flow diversion would be expected to occur during certain postulated LOCA scenarios (see Safety Significance, Section F) when the differential pressure across the spring loaded check valve exceeds the spring loading (75 psid). This flow diversion, if not discovered and corrected, could lead to a condition where containment sump inventory was insufficient to maintain continued recirculation flow.

Immediate action was taken to preclude this flow diversion by closing a manual isolation valve (VCC-317) on the TP thermal relief line.

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

None.

3. Sequence of Events:

DATE	TIME	ACTION
12/28/89	1040	A potential for diversion of recirculation flow through the spring loaded check valve was recognized.
12/28/89	1134	Manual isolation valve VCC-317 closed.

4. Method of Discovery:

Personnel discovered this condition during an evaluation of the Safety Injection System in conjunction with verifying the scope of a design change package.

5. Personnel Actions and Analysis of Actions:

Not applicable.

6. Safety System Responses:

Not applicable.

D. CAUSE OF THE EVENT:

1. Immediate cause:

Previous evaluations of recirculation flow after a LOCA, incorrectly credited continued post-accident pressurization of the VCT in order to prevent diversion of the recirculation flow through the spring loaded check valve.

2. Root cause:

The root cause of this event, as described above, and corrective actions to preclude reoccurrence are related to programmatic weaknesses described in SCE's October 3, 1988 submittal to the NRC.

E. CORRECTIVE ACTIONS:

1. Corrective Actions Taken:

To preclude any flow diversion to the VCT through this line, an isolation valve downstream of VCC-316 was closed and is being administratively controlled in the closed position. The closure of this valve prevents any diversion of recirculation flow to the VCT via the spring loaded check valve.

2. Planned Corrective Actions:

a. An engineering evaluation will be performed to determine if the spring loaded check valve will be removed or modified.

b. 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 are being implemented to address these conclusions which 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.

It is believed that the discovery of the event being reported in this LER is a positive result of these improvements in the engineering and technical support to San Onofre.

F. SAFETY SIGNIFICANCE OF THE EVENT:

The safety significance assessment of the potential flow diversion to the VCT was evaluated for two scenarios, as discussed below. For the limiting but extremely low probability event (LOCA concurrent with a design basis seismic event resulting in a loss of VCT integrity), recirculation flow to the reactor core during the recirculation phase of SI may have been inadequate for accident mitigation. For the other scenario in which VCT integrity is maintained, the amount of diversion involved is limited such that it is unlikely to result in any safety significance.

Loss of VCT Integrity:

The worst case for flow diversion would occur if VCT integrity were lost either due to a break in the tank or a connecting line or due to a non-isolable leak in a connecting line. In this case, the VCT would become depressurized and remain in that state. The pressure differential across VCC-316 would be such that for either one or two Recirculation Pumps operating, the valve would pass flow. Since the VCT would remain depressurized, the flow might be expected to continue indefinitely or until such time as it was identified and some corrective action could be taken. The expected diversion flow rate was not calculated explicitly, but it is estimated that it would be approximately 10 gpm. The results of such diversion would be as follows: 1) the available recirculation flow to the reactor vessel would be decreased, thereby reducing heat removal capability and increasing the probability of subsequent uncovering of the core; 2) continued flow diversion would lead to decreasing inventory in the containment sump. If this diversion existed for a sufficient period of time, the inventory could be reduced to the point that recirculation pump operation could not be sustained. However, at 10 gpm it would be a fairly slow process; and 3) the diverted flow would be released outside of containment resulting in increased off-site doses. Using the assumptions described in the Updated Final Safety Analysis Report (UFSAR), a release of 10 gpm of sump water to the environment would lead to thyroid doses at the Exclusion Area Boundary (0-2 hours) in excess of the limits of 10 CFR 100.

VCT Integrity Maintained:

If VCT integrity were not lost, some diversion might still occur (for the case of two recirculation pumps in operation), but the consequences would be much less severe.

In this case, total diversion would be approximately 240 gallons and would be retained in the VCT. The rate of diversion was not explicitly calculated, but would be expected to be less than the 10 gpm indicated previously and would decrease to zero as the differential pressure across the spring loaded check valve decreased to 75 psid. The consequences of this diversion would be that the recirculation flow delivered to the core would be reduced for some finite amount of time. The total diversion would be approximately 240 gallons and would be insignificant relative to

the expected sump inventory, which is in excess of 250,000 gallons. This amount of diversion at a rate of less than 10 gpm would not be expected to lead to subsequent uncovering of the core, although explicit calculations of such have not been performed. Since all of the diverted flow would be retained in the VCT, off-site doses would not be directly impacted.

Conclusions:

While the probability of a LOCA concurrent with a design basis seismic event is extremely low, it is likely that during such an event, the integrity of the VCT or connecting piping will be lost, and a flow diversion could lead to unacceptable results from inadequate core cooling or off-site doses or both. If the VCT integrity is not lost during the above event and the diverted flow is retained in the system, the consequences would be much less severe, and would most likely be shown to be acceptable.

G. ADDITIONAL INFORMATION:

1. Component Failure Information:

   Not applicable.

2. Previous LERs for Similar Events:

   The following are LERs which reported similar design and design control related conditions:

   Unit 1 (Docket No. 50-206)

   LER 86-007, Revision 1, reported a single failure susceptibility of the RPS.

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

   LER 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.

   LER 88-006, Revision 1, 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 88-009, Revision 1, reported a condition in which the emergency diesel generators could have exceeded an intended electrical load limit.

LER 88-016 reported a condition in which the south refueling water pump may not have started as required.

LER 88-017, Revision 1, reported a condition in which the auxiliary feedwater storage tank minimum volume requirements for accident mitigation may have been inadequate.

LER 88-019 reported that design deficiencies existed in automatic controls of the electrical power distribution system.

LER 88-020, Revision 1, reported that design requirements of NUREG-0737 had not been fully implemented in the design of the steam generator wide-range level indication system.

LER 89-003 reported a condition in which the failure mode of Component Cooling Water was non-conservative with respect to design requirements due to inadequate single failure analysis.

LER 89-004 reported that due to a design deficiency associated with diesel generator load sequencing logic, the automatic response capability of one of the two trains of safety-related components during a postulated SISLOP scenario may be lost.

LER 89-007 reported single failure susceptibility of the Reactor Protection System in the event of a Reactor Coolant Pump locked rotor.

LER 89-008 reported the remote possibility of a single failure which could divert cooling water from the Containment Spray System to the Containment Fire Suppression System during recirculation, thus reducing the effectiveness of the Containment Spray System.

LER 89-011, Revision 1, reported various conditions resulting in a reduction in Safety Injection delivery to the Reactor Coolant System below that assumed during certain accident scenarios.

LER 89-013 reported that the Power Operated Relief Valves were not included in the Inservice Testing program for their Overpressure Mitigation System function.

LER 89-022, Revision 1, reported that the Technical Specifications governing the Overpressure Mitigation System permitted non-conservative plant operation.

LER 89-024 reported a condition that could result in a reduction in containment spray flow thereby, reducing the effectiveness of the Containment Spray System.

LER 89-025 reported a condition involving the susceptibility of Hot Leg Recirculation to a loss of non-safety related instrument air.

LER 89-029 reported the potential for a flow diversion from the Refueling Water Storage Tank (RWST) through the RWST Recirculation Filter following a postulated seismic event.

Unit 2 (Docket No. 50-361)

LER 88-008, Revision 1, reported various conditions resulting in the component cooling water system being outside its design basis due to design control program deficiencies.

LER 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 88-017, Revision 1, reported that a spent fuel pool siphon event occurred as a result of the failure to transfer the design intent to utilize administrative controls on certain locked valves.

LER 88-034 reported a condition involving safety related Component Cooling Water System valves being susceptible to seismically-induced common mode failures.

LER 89-007 reported that the design requirements of the DG Starting Air Systems (SAS) may not have been adequately demonstrated during startup testing.

LER 89-012, Revision 1, reported instrument mounting configuration discrepancies caused by Environmental Qualification and design control program weaknesses.

3. Results of NPRDS Search:

Not applicable.