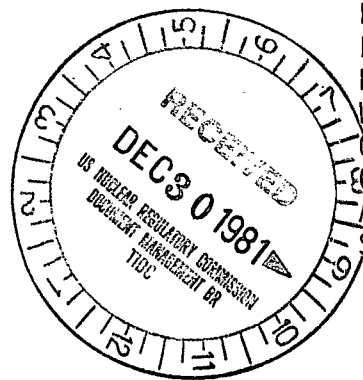


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December 28, 1981

Docket No. 50-206
 LS05-81-12-079



Mr. R. Dietch
 Vice President
 Nuclear Engineering and Operations
 Southern California Edison Company
 2244 Walnut Grove Avenue
 Post Office Box 800
 Rosemead, California 91770

Dear Mr. Dietch:

SUBJECT: REACTOR VESSEL OVERPRESSURE PROTECTION - SAN ONOFRE UNIT 1

We are continuing our review of the multi-plant issue of reactor vessel overpressure protection for San Onofre Unit 1 and find that additional information described in the enclosure to this letter is needed. We request your response by February 15, 1982.

The reporting and/or recordkeeping requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511.

Sincerely,
 Original signed by
 Walter A. Paulson for/
 Dennis M. Crutchfield, Chief
 Operating Reactors Branch #5
 Division of Licensing

Enclosure:
 Request for Additional
 Information

cc w/enclosure:
 See next page

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 PDR

OFFICE	DL: ORB #5	DL: ORB #5	DL: AD/SA				
SURNAME	WPaulson	cc: DCrutchfield	GLamas				
DATE	12-23-81	12/28/81	12/28/81				

December 28, 1981

cc

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Mayor
City of San Clemente
San Clemente, California 92672

Chairman
Board of Supervisors
County of San Diego
San Diego, California 92101

California Department of Health
ATTN: Chief, Environmental
Radiation Control Unit
Radiological Health Section
714 P Street, Room 498
Sacramento, California 95814

U. S. Environmental Protection Agency
Region IX Office
ATTN: Regional Radiation Representative
215 Fremont Street
San Francisco, California 94111

1. The Branch Technical Position, RSB 5-2, requires the overpressure mitigation systems (OMS) to meet single active failure analysis when the initiating cause of the event is not considered as the single active failure. In your analysis submitted in your October 12, 1977 letter, you cover many single failures of system active components. Address the following scenario for the San Onofre OMS design.

Consider as the initiating event the failure of a vital bus which results in the isolation of letdown flow (i.e., fails closed PCV-1105 or TCV-1105 and LCV-1100A) and also fails closed one of the pressurizer PORVs. A postulated single failure (failing closed) of the other PORV would fail mitigating systems for the event. Discuss your plant's provisions to mitigate such a transient.

2. The branch position requires an alarm to alert the operator to enable the OMS at the correct plant condition during cooldown. You rely on a pressure actuated alarm to perform this function. How do you ensure that the Reactor Coolant System temperature does not fall below the allowable temperature corresponding to the above alarm pressure setpoint, thus violating limits specified in Appendix G to 10 CFR Part 50.
3. You state that the required functional test will be performed prior to returning to a water-solid condition following a cold shutdown. Will this test also be performed prior to placing the OMS in service during plant cooldown from a long period of operation? If not, give your basis for not testing the OMS at this time.
4. The Branch Technical Position requires the OMS to function during an Operating Basis Earthquake. You stated that portions of your OMS meet Seismic Category A requirements and that the rest of the system meets Category B.

- a. Identify all system components that are not designed to function during an Operating Basis Earthquake, and give the basis for these components not meeting this requirement.
 - b. Analyze the situation if the components that you identify in 'a' above do not function.
 - c. Are the PORV operators qualified to operate through an OBE?
5. Additional information is requested on the following items concerning RCS and RHR relief and safety valves.
- a. Is the relief capacity of RHR safety valve RV-206 sufficient to mitigate the overpressure transients that were analyzed for the OMS?
 - b. Does the valve position indication of the PORVs and the PORV block valves indicate a control circuit condition or the actual position of the valves?
6. In your May 2, 1977, you provided some information on the training that you conducted on the overpressurization incidents; provide the following additional training information.
- a. What overpressure training have you performed since 1977?
 - b. How do you ensure that a continued emphasis is placed on possible overpressurization situations in your licensing and retraining programs?
 - c. How is this training and LER review documented?
7. List all the administrative procedures and controls used to minimize the probability of an overpressure transient (i.e., solid-water operation limited to certain RCS pressure, pressurizer heater and HPI pumps disabled at certain RCS pressure and temperature, etc.) Indicate which of the administrative procedures and controls is in the plant's Technical Specifications.

8. What is the present status of the San Onofre-1 OMS?
 - a. Have all permanent OMS installations and modifications been completed?
 - b. Have warnings and caution notes been included in all affected procedures?
 - c. Have all necessary administrative documentation changes been made?

9. If the Westinghouse generic analysis applies to your plant, reactor coolant pumps should not be started when water-solid and with a temperature difference of $\geq 50^{\circ}\text{F}$ between the RCS and the steam generator secondary side. What means are available at San Onofre-1 to determine the representative temperature difference between the RCS and the steam generators?

10. During the 'no-flow test' portion of the cold operation test of the safety injection system while the RCS is water-solid, is the OMS lined up to provide RCS protection? If the answer above is no, please discuss the following scenarios: If an operator error resulted in a feedwater pump breaker not being racked out; would starting a feedwater pump, with only the water in the piping as a source (suction valves remain closed), result in violating the Appendix G limits?

11. Have you continued the practice of assigning a reactor operator to monitor reactor coolant system pressure during all water-solid operations?

12. What is the reactor pressure vessel age in effective full power years (EFPY) for which your Appendix G limits are calculated?