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MANAGER OF NUCLEAR ENGINEERING,
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(213) 572-1401

February 23, 1982

Director, Office of Nuclear Reactor Regulation
Attention: D. M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

Subject: Docket No. 50-206
Reactor Vessel Overpressure Protection
San Onofre Nuclear Generating Station
Unit 1

Your December 28, 1981 letter requested additional information on the issue of reactor vessel overpressure protection for San Onofre Unit 1. We indicated in our February 10, 1982 letter that we would provide the requested information by February 24, 1982. Enclosed please find our response to your questions.

If you have any questions concerning this information, please contact me.

Very truly yours,

K P Baskin

Enclosure

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Enclosure

Additional Information on Reactor Vessel Overpressure Protection
San Onofre Unit 1

These answers correspond to questions raised in the December 28, 1981 letter from D. M. Crutchfield, NRC to R. Dietch, SCE.

1. Question

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.

Response

As stated in Reference 1, the RHR system is put into service before the bubble is collapsed and not taken out of service until after the bubble is re-established. Therefore, anytime the RCS is water solid, the RHR system is open (MOV 813 and MOV 814). MOV-813 and MOV-814 do not close automatically on high pressure in the RCS. Thus, RV-206 would be available to mitigate a pressure transient resulting from isolation of the normal let down path. The relief capacity of RV-206 is sufficient to mitigate the overpressure transients analyzed for the OMS as indicated in the response to 5a.

2. Question

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?

Response

Plant cooldown is accomplished utilizing operating instruction SO1-3-5, "Plant Shutdown From Hot Standby to Cold Shutdown." Adherence to this instruction will ensure RCS cooldown complies with Technical Specification 3.1.3. This Technical Specification requires RCS temperature to be maintained in excess of that required to meet the limits specified in Appendix G to 10 CFR Part 50.

3. Question

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.

Response

Functional testing is not performed prior to placing the OMS in service during plant cooldown from a long period of operation. Placing the system in service at this time would increase the probability of a loss-of-coolant due to PORV malfunction. By not placing the system in operation, the PORVs will not be required to open and thus the possibility that they would fail in an open position or seat improperly is eliminated.

4. Question

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 those 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?

Response

a. The following portions of the air supply system to the PORV's are Non-Seismic Category A:

Pneumatic Tubing
Tube Fittings
In-Line Check Valves
Branch Line Globe Valves
In-Line Globe Valves
Support Hardware (except where failure could lead to damage to seismic A equipment)

The backup nitrogen system is Seismic Category A except for sections that remain as original plant equipment. The system seismic design is based on the extremely low probability of experiencing a DBE during water solid operation.

b. Failure of the above components would disable the OMS.

c. The PORV operators are designed as seismic category A as stated in the FSA.

5. Question

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?

Response to 5a

The relief capacity of RHR safety valve RV-206 is sufficient to mitigate the overpressure transients that were analyzed for the OMS. The setpoint for RV-206 is 500 psig. The capacity of RV-206 is 783 gpm at 25% accumulation (625 psig). This is equivalent to the capacity of 1 PORV at the same pressure. At 10% accumulation (550 psig) the capacity of RV-206 is 470 gpm. A review of the overpressure transients analyzed for the OMS indicated this capacity is sufficient to turnaround the pressure transient, and the corresponding pressure is below the Appendix G Technical Specification limit (555 psig) for the limiting overpressure transient.

Response to 5b

The valve position indication of the PORVs and PORV block valves indicate the actual position of the valve via limit switches on the valve stem.

6. Question

In your May 2, 1977 letter, 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?

Response

a. A review of annual requalification exams for 1977 and 1978 revealed a question on the 1978 exam regarding OMS operation. Recent discussions with operators on the staff indicate special training was given in 1977 and 1978 though no documentation of this is available.

Documented training since 1979 includes the following with regard to OMS:

The July, 1979 retraining lecture on safety and emergency systems covered the OMS.

The June, 1980 off shift training reviewed selected emergency operating instructions including instruction on OMS operation.

The July, 1981 on shift reading assignment included the emergency operating instruction on OMS actuation.

b. Continued emphasis is placed on possible overpressurization situations through a continuation of our procedure review program and through monthly off-shift lectures. In addition, cautionary notes are included in applicable procedures.

c. Training and review of documents of significant impact on station operation (e.g., changes in procedures, significant operating events, LERs) are done through required reading of documents and monthly off-shift lectures. Each operator is required to sign an acknowledgement of information card which indicates that he has read all significant documents. The compliance group then verifies monthly that this review has occurred by checking each operator against the list of documents to be reviewed.

7. Question

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.

Response

The following operating instructions provide guidance to minimize the probability of an overpressure transient.

SO1-1.2-13	Overpressurization Mitigating System (OMS) Actuation
SO1-3-1	Plant Startup from Cold Shutdown to Hot Standby

SO1-3-5	Plant Shutdown from Hot Standby to Cold Shutdown
SO1-4-1	Filling and Venting the Reactor Coolant System
SO1-4-4	Reactor Coolant Pump Startup
SO1-4-9	Placing the Residual Heat Removal System In Service
SO1-4-17	Safety Injection System Operations

Technical Specification 3.3.1 "Combined Heatup, Cooldown and Pressure Limitations" is a reference for the preceeding operating instructions. In addition, Technical Specification 3.3.2 "Shutdown Status" requires two positive barriers between the feedwater condensate system and the piping to the RCS when RCS pressure is below 500 psi.

8. Question

What is the present status of the San Onofre-1 OMS?

a. Have all the 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?

Response

The present status of SONGS 1 Overpressure Mitigation System is complete and operational. All appropriate warnings and cautions have been incorporated in the affected operating instructions and the necessary documentation changes have been made.

9. Question

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 $\geq 50^{o}F$ 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?

Response

The Westinghouse generic analysis does apply to SONGS-1. A representative temperature difference between the RCS and the steam generator secondary is determined from the average RCS temperature and the saturation temperature of the steam generator secondary side. The RCS average temperature is measured by RTD's (Resistance Temperature Detectors) located in the RCS hot and cold legs. The saturation temperature of the steam generator is inferred from a pressure indicator connected to the steam generator secondary side. Operating Instruction SO1-3-5 indicates that reactor coolant pumps may not be restarted unless RCS temperature gradients are evaluated.

10. Question

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?

Response

The OMS is lined up to provide RCS protection during the "no-flow test" of the SIS. As indicated in Reference 2, this test is not performed with the RCS water-solid and the potential for an inadvertent SIS actuation is remote. In addition, 2 positive barriers are maintained between the feedwater condensate system and the RCS per Technical Specification 3.3.2.

11. Question

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

Response

No. Assigning of an operator to monitor RCS pressure during water-solid operations is only required when the OMS is inoperative.

12. Question

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

Response

The current Appendix G operational limits are based on 16 equivalent full power years (EFPY).

References

1. Letter, K. P. Baskin, SCE, to A. Schwencer, NRC, Evaluation of Reactor Vessel Overpressurization Susceptibility, October 12, 1977.

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