July 22, 1982

Docket No. 50-206
LS05-82- 07-061

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: SEP TOPIC XV-1, DECREASE IN FEEDWATER TEMPERATURE, INCREASE IN FEEDWATER FLOW, INCREASE IN STEAM FLOW, AND INADVERTENT OPENINGS OF A STEAM GENERATOR RELIEF OR SAFETY VALVE

By letter dated October 22, 1981, the staff issued a safety evaluation on SEP Topic XV-1. Your letter of March 12, 1982, provided comments on the staff's evaluation. Enclosed is the staff's final topic evaluation. The staff's position is that you should demonstrate that a continued increase in feedwater flow event, using a minimum time of ten minutes for operator action, would not result in violation of acceptance criteria for fuel thermal limits and design pressures.

As previously noted the subject of steam generator overfill from control system failures will be included in the scope of Unresolved Safety Issue A-47; however, the schedule for completion of Task A-47 is not compatible with the SEP Integrated Assessment schedule.

The enclosed final safety evaluation will be a basic input to the Integrated Safety Assessment for your facility. The assessment may be revised in the future if your facility design is changed or if NRC criteria relating to this topic are modified before the Integrated Assessment is completed.

Sincerely,

SE04
DSU USE EX(8)

Original signed by:
Walter A. Paulson, Project Manager
Operating Reactors Branch No. 5
Division of Licensing

Add: Gary Staley

Enclosure:
As stated

8207300264 820722
PDR ADOCK 05000206
P PDR

cc w/enclosure:
See next page

OFFICE	SEPB	SEPB	SEPB	ORB#5	ORB#5	AD:SA
SURNAME	EMcKenna:bl	CGrimes	WRussell	WPaulson	DCrutchfield	GLainas
DATE	7/19/82	7/19/82	7/19/82	7/20/82	7/22/82	7/21/82

cc
Charles R. Kocher, Assistant
General Counsel
James Beoletto, Esquire
Southern California Edison Company
Post Office Box 800
Rosemead, California 91770

David R. Pigott
Orrick, Herrington & Sutcliffe
600 Montgomery Street
San Francisco, California 94111

Harry B. Stoehr
San Diego Gas & Electric Company
P. O. Box 1831
San Diego, California 92112

Resident Inspector/San Onofre NPS
c/o U. S. NRC
P. O. Box 4329
San Clemente, California 92672

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 Freemont Street
San Francisco, California 94111

Robert H. Engelken, Regional Administrator
Nuclear Regulatory Commission, Region V
1450 Maria Lane
Walnut Creek, California 94596

SYSTEMATIC EVALUATION PROGRAM
TOPIC XV-1

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TOPIC XV-1: DECREASE IN FEEDWATER TEMPERATURE, INCREASE IN FEEDWATER FLOW, INCREASE IN STEAM FLOW AND INADVERTENT OPENING OF A STEAM GENERATOR RELIEF OR SAFETY VALVE

I. INTRODUCTION

These events involve an unplanned increase in heat removal by the secondary system which can cause a decrease in the temperature of the reactor coolant, an increase in reactor power due to the negative moderator temperature coefficient and a decrease in the reactor coolant system steam generator pressure.

Depending on the magnitude of the temperature reduction the plant may stabilize in new operating conditions, or the overpower or variable low pressure protection may cause a reactor trip. Each event description below has a separate section for evaluation and conclusions.

II. REVIEW CRITERIA

Section 50.34 of 10 CFR Part 50 requires that each applicant for a construction permit or operating license provide an analysis and evaluation of the design and performance of structures, systems and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility, including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility.

The General Design Criteria (Appendix A to 10 CFR Part 50) establish minimum requirements for the principal design criteria for water-coolant reactors. GDC 10 "Reactor Design" requires that the core and associated coolant, control and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operation, including the effects of anticipated operational occurrence.

GDC 15 "Reactor Coolant System Design" requires that the reactor coolant and associated protection systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during normal operation, including the effects of anticipated operational occurrences.

GDC 26 "Reactivity Control, System Redundancy and Capability" requires that the reactivity control systems be capable of reliably controlling reactivity changes to assure that under conditions of normal operation including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded.

.II. RELATED SAFETY TOPICS

Various other SEP topics evaluate such items as the reactor protection system. The effects of single failures on safe shutdown capability are considered under Topic VII-3.

IV. REVIEW GUIDELINES

The review is conducted in accordance with SRP 15.1.1., 15.1.2, 15.1.3 and 15.1.4.

The evaluation includes review of the analysis for the event and identification of the features in the plant that mitigate the consequences of the event as well as the ability of these systems to function as required. The extent to which operator action is required is also evaluated. (Draft Standard ANS 58.8 used as guidance). Deviations from the criteria specified in the Standard Review Plan are identified.

V. INDIVIDUAL EVENT REVIEW

DECREASE IN FEEDWATER FLOW

A. Evaluation

The licensee has not presented a detailed analysis of the transient, but has identified the increase in feedwater flow transient as a more limiting bounding event.

The feedwater system consists of two trains, each having one high pressure heater and a set of low pressure heaters. From the figure in Reference 1, which presents a heat balance of the turbine cycle, it can be seen that the high pressure heaters increase the enthalpy by 301.0 Btu/lb. Each of these heat sources could be bypassed by opening a single valve in a bypass line.

The licensee has estimated the consequences of the loss of both high pressure heaters in Reference 1, but has not considered the possibility of the loss of the whole set of low pressure heaters. However, from the plant energy balance, it is estimated that the increase in feedwater flow transient is also a bounding event for loss of all low pressure heaters in one train.

B. Conclusion

It is concluded that the decrease in feedwater temperature transient is bounded by the increase in feedwater flow transient.

INCREASE IN FEEDWATER FLOW

A. Evaluation

An increase in feedwater flow can result from excessive opening of the feedwater control valve, overspeed of a feedwater pump, or starting a second feedwater pump. The increase in feedwater flow will cause more heat to be extracted by the steam generator. When colder water reaches the core, the power is increased through negative reactivity feedback.

The licensee has presented an analysis of increase in feedwater flow in Reference 2 and 3.

The largest increase in feedwater flow at full load would be caused by simultaneous opening of all three feedwater control valves. Feedwater flow is limited by the feedwater system head-flow characteristics. The licensee's analysis has conservatively assumed a step increase to 140% of full flow, which is beyond the actual limit of the feedwater system. However, in order to limit the consequences of this transient the operator is assumed to manually trip the reactor within two minutes following the increase in feedwater.

As a second case, the licensee has analyzed a step increase from 43% to 103% feed flow when operating at a steady-state power of 51% of design. A conservative low feedwater temperature is assumed for this case. For this case the operator is assumed to manually trip the reactor within 40 seconds following this increase in feedwater.

All core parameters used in the analysis are more conservative than the actual parameters calculated for the current cycle. No automatic plant controls are assumed to function, but a manual reactor trip is assumed soon after the operator has been alerted by high level alarms on the steam generator. The other assumptions made in the analysis are in conformance with the criteria of SRP Section 15.1.1.

The results of the analysis show that in both cases the primary system parameters smoothly approach their asymptotic values and do not reach the protective limits. Thus there is not immediate concern of DNB or overpressure.

Even if credit is not taken for manual reactor trip, the automatic reactor protection system features would initiate a reactor trip if DNB or pressure limits were approached.

Upon detection of high steam generator water level, the water level control system would attempt to close the feedwater regulating valves. High steam generator water level will also initiate a turbine trip, which, with power greater than 10% of rated, will trip the reactor.

Main feedwater addition to the generator could be terminated by closure of the regulating valves (control system or manual), by closure of the block valves (manual), or by tripping of the main feedwater pumps (upon safety injection actuation, if reached, or manual).

The licensee has not established that the time available for termination, by the operator, of feedwater addition prior to flooding the steam lines is in accordance with current guidelines. Further, the consequences of not terminating the increase in feedwater flow transient before the steam lines have flooded have not been evaluated. If the flooding would result in a steamline rupture the ensuing reactor coolant system overcooling could be more severe than during the worst case steamline break accident analyzed by the licensee.

It is our position that the event should be reanalyzed using the Draft ANSI Standard N660 guidance for operator action. The same issue (steam generator overfill) is within the scope of Task A-47, Safety Implications of Control Systems, of the NRC program of Unresolved Safety Issues.

B. Conclusions

The analysis of the increase in feedwater flow event has been evaluated against the criteria of SRP 15.1.1. We have concluded that the operator action times assumed in the analyses do not meet the criteria of ANSI N660 and that credit for such times should not be given. The licensee should demonstrate that an increase in feedwater flow event, using a minimum time of ten minutes for operator action, would not result in violation of the acceptance criteria for fuel thermal limits and design pressures.

INCREASE IN STEAM FLOW

A. An increase in steam flow may be initated by opening of the turbine control valves. The plant response to steam flow increase depends on the control mode and the magnitude of the moderator reactivity coefficient. In general, however, the core power tends to increase to a level matching the increased steam flow. In the automatic control mode, the power is increased by a combined effect of the control rod withdrawal and the feedback from moderator temperature. In the manual control mode, it is increased as a result of the negative moderator temperature coefficient when the primary temperature and both primary and secondary pressures decrease.

The licensee has presented an analysis of an increase in steam flow in Reference 2.

The maximum thermal power level for the current operating conditions, evaluated in Reference 4, corresponds to the situation with the turbine control valves fully open. This eliminates the possibility of a load increase above this power level.

The event analyzed in Reference 2 is a transient response to a sudden requirement for 30 percent more load by the turbine governor control while operating at 70 percent load. An automatic reactor control and a slightly positive moderator coefficient are assumed which yield the most severe transient. Manual control or negative moderator coefficient would result in a more smooth transfer to a new equilibrium state.

The results of the analysis show that the plant parameters do not reach the protection limits, set to protect the plant against DNB and overpressure.

B. Conclusion

The analysis of increase in steam flow has been evaluated against the criteria of SRP 15.1.1 and we have concluded that it is in conformance with the criteria.

INADVERTENT OPENING OF A STEAM GENERATOR RELIEF OR SAFETY VALVE

A. Evaluation

An atmospheric dump valve may be inadvertently opened by the operator or may open due to a failure in the control system that opens the valve. A steam generator safety valve may be opened only as a result of a valve failure. The consequences of an inadvertent opening of either valve are the same as for a small break in the steam line: a reduction of the reactor coolant temperature and pressure and insertion of reactivity because of a negative moderator temperature coefficient.

If the reactor is at power during the inadvertent opening of a relief or safety valve, a reactor trip is caused by overpower protection.

Continued cooling to hot shutdown conditions actuates the safety injection system due to coincident low pressurizer pressure and low pressurizer level signals. The core is ultimately shut down by the boric acid injection delivered by the safety injection system.

The licensee has analyzed inadvertent opening of a steam generator relief or safety valve several times in the past. Results of the most recent analysis are presented in Reference 4.

Inadvertent opening of a valve, equivalent to a steam leak of 152 lb/sec at 920 psia, has been analyzed assuming initial hot shutdown conditions. This causes a larger and more rapid cooling than an event starting from power operation, because the stored energy in the reactor coolant system is smaller and the water inventory and pressure in the steam generator are higher.

The method and the assumptions used in the analysis are in conformance with the criteria of SRP Section 15.1.1. The core parameters are more conservative than the actual parameters calculated for the current cycle.

The results of the analysis show that the reactor does not become critical during the transient. Thus there is no concern of DNB or overpressure.

B. Conclusion

The analysis of an inadvertent opening of a steam generator relief or safety valve has been evaluated against the criteria of SRP 15.1.1 and we have concluded that it is in conformance with the criteria.

VI. TOPIC CONCLUSIONS

Each of the events covered by this topic have been reviewed and we have concluded that the analyses are in conformance with SRP criteria with the following exception:

The licensee should demonstrate that an increase in feedwater flow event, using a minimum time of ten minutes for operator action, would not result in violation of the acceptance criteria for fuel thermal limits and design pressures.

REFERENCES

1. Letter, W. C. Moody (SCE) to D. M. Crutchfield (NRC) Subject: Design Basis Event Reviews, Systematic Evaluation Program, San Onofre Nuclear Generating Station Unit 1, July 1, 1981.

2. San Onofre Nuclear Generating Station Unit 1, Part II Final Safety Analysis, 1970.

3. San Onofre Nuclear Generating Station Unit 1, Part I Operating History and Verification of Design Objectives, Appendix A, 1970.

4. Reload Safety Evaluation Cycle 8, Revision 1, San Onofre Nuclear Generating Station Unit 1, October 1980.