November 3, 1982

Docket No. 50-206
LS05-82-11-007

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-7, REACTOR COOLANT PUMP ROTOR SEIZURE AND REACTOR COOLANT PUMP SHAFT BREAK SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 1

In your letter dated July 1, 1981, you submitted a safety assessment on the above topic. The staff has reviewed your assessment and our conclusions are presented in the enclosed safety evaluation report. As noted in the evaluation, it is the staff's position that you should demonstrate that the consequences of decrease in reactor coolant flow events satisfy the applicable review criteria.

The actions necessary to resolve this concern will be addressed in the integrated plant safety assessment. 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 you facility design is changed or if NRC criteria relating to this topic are modified before the integrated assessment is completed.

Sincerely,

Original signed by:

Walter Paulson, Project Manager
Operating Reactors Branch No. 5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

SE04

DSU USE EX(08)

Add: Gary Staley

8211080163 821103
PDR ADOCK 05000206
P PDR

OFFICE	SEPB EM	SEPB CG	SEPB	ORB#5	ORB#5	AD:SA:DL	
SURNAME	EMcKenna:bl	CGrimes	WRussell	WPaulson	DCrutchfield	FMiraglia	
DATE	10/21/82	10/ /82	10/ /82	10/ /82	10/ /82	10/ /82	

Mr. R. Dietch

San Onofre Unit 1
Docket No. 50-206
Revised 3/30/82

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

SEP TOPIC XV-7(a)

SAN ONOFRE 1

SUBJECT: LOSS OF FORCED REACTOR COOLANT FLOW INCLUDING TRIP OF PUMP MOTOR AND FLOW CONTROLLER MALFUNCTIONS

I. Introduction

A decrease in reactor coolant flow occurring while the plant is at power could result in a degradation of core heat transfer. A resulting increase in fuel temperature and accompanying fuel damage could then result, if specified acceptable fuel design limits are exceeded during the transient. A number of transients that are expected to occur with moderate frequency and that result in a decrease in forced reactor coolant flow rate are addressed in SRP 15.3.1 and SRP 15.3.2.

II. Review Criteria

Section 50.34 of 10 CFR Part 50 requires that each applicant for an operating license provide an analysis and evaluation of the design and performance of structures, systems, and components of facility with the objectives of assessing the risk to public health and safety resulting from operation of the facility. The loss of forced reactor coolant flow is one of the postulated transients used to evaluate the adequacy of these structures, systems and components with respect to the public health and safety.

Section 50.36 of 10 CFR Part 50 requires the Technical Specifications to include safety limits which protects the integrity of the physical barriers which guard against the uncontrolled release of radioactivity.

The General Design Criteria (Appendix A to 10 CFR Part 50) establish minimum requirements for the principal design criteria for water-cooled reactors. The staff acceptance criteria are based on meeting the relevant requirements of the following regulations:

A. General Design Criterion 10, as it relates to the reactor coolant system being designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operations including anticipated operational occurrences.

B. General Design Criterion 15, as it relates to the reactor coolant system and its associated auxiliaries being designed with appropriate margin to assure that the pressure boundary will not be breeched during normal operations including anticipated operational occurrences.

C. General Design Criteria 26, as it relates to the reliable control of reactivity changes to assure that specified acceptable fuel design limits are not exceeded, including anticipated operational occurrences. This is accomplished by assuring that appropriate margin for malfunctions, such as stuck rods, are accounted for.

III. Related Safety Topics

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

IV. Review Guidelines

The review is conducted in accordance with SRP Sections 15.3.1 and 15.3.2. The evaluation includes reviews 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. Deviations from the criteria specified in the Standard Review Plan are identified.

The specific criteria identified in the SRP as necessary to meet the relevant requirements of GDC 10, 15 and 26 for incidents of moderate frequency are:

a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values.

b. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs and the CPR remains above the MCPR safety limit for BWRs based on acceptable correlations (see SRP Section 4.4).

c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

d. An incident of moderate frequency in combination with any single component failure, or single operator error, shall be considered and is an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. For such accidents, the number of fuel failures assumed must be equal to the number of all rods for which the DNBR or CPR falls below those values cited above for cladding integrity unless it can be shown, based on an acceptable fuel damage model (see SRP Section 4.2), that fewer failures occur. There shall be no loss of function of any fission product barrier other than the fuel cladding.

V. Evaluation

The licensee has not updated his analysis of the loss of coolant flow event, as requested by the staff, in order to meet the requirements of the SEP program. However, analyses performed earlier by the licensee in the FSAR (Ref. 1) have assumed loss of all reactor coolant pumps and a 90 percent and a 85 percent low flow reactor trip at an initial power level of 103 percent.

The results of the FSAR analyses determined a minimum DNBR of 1.62 for the 90 percent reactor trip case and a minimum DNBR of 1.43 for the 85 percent reactor trip case. The peak reactor coolant pressure is not reported by the licensee for these analyses.

VI. Conclusion

Based on our evaluation of the licensee's analyses of the loss of coolant flow event, we have concluded that an updated analysis should not affect minimum DNBR. However, we are unable to conclude whether the peak reactor coolant pressure during this event would exceed the allowable limit (110 percent of the design pressure). We, therefore, recommend that the consequences of this event and the need for additional analyses be reviewed during the plant integrated assessment.

VII. Reference

1. Southern California Edison Co., San Onofre Nuclear Generating Station, Unit 1, Final Safety Analysis Report, Part II, Vol. V.

SEP TOPIC XV-7 (b)

SUBJECT: REACTOR COOLANT PUMP ROTOR SEIZURE AND REACTOR COOLANT PUMP SHAFT BREAK

I. Introduction

The events postulated are an instantaneous seizure of the rotor or break of the shaft of a reactor coolant pump. Flow through the affected loop is rapidly reduced. The sudden decrease in core coolant flow while the reactor is at power results in a degradation of core heat transfer which could result in fuel damage. The initial rate of reduction of coolant flow is greater for the rotor seizure event. However, the shaft break event permits a greater reverse flow through the affected loop later during the transient and, therefore, results in a lower core flow rate This topic is intended to cover both of these accidents.

II. Review Criteria

Section 50.34 of 10 CFR Part 50 requires that each applicant for an 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. The reactor coolant pump rotor seizure and reactor coolant pump shaft break are two of the postulated accidents used to evaluate the adequacy of these structures, systems, and components with respect to the public health and safety.

Section 50.36 of 10 CFR Part 50 requires the Technical Specifications to include safety limits which protect the integrity of the physical barriers which guard against the uncontrolled release of radioactivity.

The General Design Criteria (Appendix A to 10 CFR Part 50) establish minimum requirements for the principal design criteria for water-cooled reactors.

GDC 27 "Combined Reactivity Control System Capability," requires that the reactivity control systems, in conjunction with poison addition by the emergency core cooling system, has the capability to reliably control reactivity changes to assure that under postulated accident conditions, and with appropriate margin for stuck rods, the capability to cool the core is maintained.

GDC 28 "Reactivity Limits" requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core.

GDC 31 "Fracture Prevention of Reactor Coolant Pressure Boundary" requires that the boundary be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fractures is minimized.

10 CFR Part 100.11 provides dose guidelines for reactor siting against which calculated accident dose consequences may be compared.

III. Related Safety Topics

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

IV. Review Guidelines

The review is conducted in accordance with SRP 15.3.3, 15.3.4. 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. Deviations from the criteria specified in the Standard Review Plan are identified.

V. Evaluation

The licensee has not submitted analyses for the reactor coolant pump rotor seizure and pump shaft break events and these events were not previously analyzed in the plant FSAR. The rotor seizure/shaft break accident results in a rapid cessation of flow from the affected pump. A reactor trip on low coolant flow would occur. Upon reactor trip, the other two reactor coolant pumps would trip and coastdown. The licensee should demonstrate that the consequences of this event satisfy the topic review criteria or are acceptable on another basis.

VI. Conclusion

We are unable to determine whether the consequences of the reactor coolant pump rotor seizure or pump shaft break event would satisfy our acceptance criteria. We, therefore, recommend that the consequences of these events and the need for additional analysis be reviewed during the plant integrated assessment.