March 03, 1982

Docket No. 50-206
LS05-82-03-012

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: SAN ONOFRE - SEP TOPIC XV-6, FEEDWATER SYSTEM PIPE BREAKS INSIDE AND OUTSIDE CONTAINMENT (PWR)

By letter dated July 1, 1981, you submitted a safety assessment report for the above topic. The staff has reviewed this assessment and our conclusions are presented in the enclosed Safety Evaluation Report, which completes the review of this topic for San Onofre Unit 1.

This evaluation will be a basic input to the integrated assessment for your facility. The evaluation 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,

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

SEO4 s 1/1 DSU WE(08)

Enclosure:
As stated

cc w/enclosure:
See next page

OFFICE	SEPB:DL	SEPB:DL	SEPB:DL	ORB#5:PM	ORB#5:BC	AD:SA:DL	
SURNAME	EMcKenna:dk	GCwalina	WRussell	WPaulson	DCrutchfield	GLainas	
	2/25/82	2/25/82	2/25/82	3/3/82	3/1/82	3/1/82	

Mr. R. Dietch

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
6600 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

Mission Viejo Branch Library
24851 Chrisanta Drive
Mission Viejo, California 92676

Mayor
City of San Clemente
SSan 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
Office of Inspection and Enforcement
1450 Maria Lane
Walnut Creek, California 94596

SEP TOPIC XV-6

San Onofre Nuclear Generating Station Unit 1

Subject: Feedwater System Pipe Breaks Inside and Outside Containment (PWR)

I. INTRODUCTION

A feedwater line break, on the pump side of the main feedwater check valve, will result in the loss of main feedwater flow to all steam generators. The feedwater line break upstream of the check valve is bounded by the analysis of loss of normal feedwater flow (SEP Topic XV-5).

A feedwater line break between the main feedwater line check valve and the steam generator will result in the complete blowdown of one steam generator. Furthermore, a break on the steam generator side may prevent or reduce the subsequent addition of auxiliary feedwater to the steam generators.

A feedwater line break can result in either a reactor system cooldown (by excessive energy discharge through the break such as that from a steamline break) or a reactor coolant system (RCS) heatup (by reducing feedwater flow to the affected steam generator). Potential RCS cooldown resulting from a secondary pipe break is bounded by the analysis of main steamline break. Therefore, only the RCS heatup effects are evaluated for a feedwater line rupture.

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.

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.

GDC 35 "Emergency Core Cooling" requires that a system be provided to provide abundant emergency core cooling whose function is to transfer heat from the core following a loss of coolant such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented, and (2) clad metal water reaction is limited to negligible amounts. The system should have suitable redundancy and interconnections such that system function can be maintained assuming a single failure and assuming available of only on-site or only off-site power supplies.

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

III. RELATED SAFETY TOPICS

SEP Topics III-5.A "Effects of Pipe Break on Structures, Systems and Components Inside Containment" and III-5.B "Pipe Break Outside Containment" consider the dynamic effects (pipe whip, jet impingement, adverse environment) on safety-related equipment. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," Task II.E.1 addresses the issue of improving the reliability of the auxiliary feedwater system. Other SEP topics address such items as ESF initiation, the auxiliary feedwater system capacity, and containment isolation.

IV. REVIEW GUIDELINES

The review is conducted in accordance with SRP 15.2.8. The evaluation includes review of the analysis of the event. 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. Deviations from the criteria specified in the SRP are identified.

V. EVALUATION

The licensee has performed an analysis (References 1 and 2) to demonstrate that the system is capable of sustaining a feedwater line rupture under initial

conditions and assumptions which result in the most severe heatup of the primary system. A detailed analysis using the LOFTRAN code was performed in order to determine the plant transient following a feedline rupture. One case was analyzed with the two intact steam generators each receiving 150 gpm of auxiliary feedwater or a total of 300 gpm at 15 minutes following the feedline break. Major assumptions for this case are as follows:

1. 103% initial power.
2. Main feedwater to all steam generators assumed to stop at the time the break occurs i.e. all main feedwater spills out through the break.
3. A saturated steam blowdown for the two intact steam generators.
4. No credit taken for the Pressurizer PORV or pressurizer spray.
5. React or trip assumed to be initiated with the steam flow/feedwater flow mismatch trip 5 seconds into the transient allowing time for signal generation and processing.
6. Operator action assumed at 10 minutes to isolate auxiliary feedwater to the affected steam generator and align the system to deliver 150 gpm flow to the two intact steam generators. A 5 minute additional delay time is assumed for refill of the main feedlines to the two intact steam generators between the check valve and steam generator nozzle.
7. Operation of the reactor coolant pump assumed throughout the transient.
8. No single failure assumed, i.e., the motor-driven auxiliary pump is assumed to be operable (the steam supply to the turbine-driven pump is lost for secondary break events).

Since San Onofre Unit 1 has no main steamline stop valves, all three steam generators will blowdown through the break. Calculated plant parameters following a major feedwater line rupture for the case analyzed are presented in Reference 2.

A heat-up is observed during the first 15 seconds of the transient due to an apparent reduction in load with the loss of subcooled main feedwater combined with a stored heat increase from a delayed reactor trip (occuring 5 seconds after the break). The reactor trip and a rapid decrease in the level of decay

heat greatly reduce the heat generation rate. Sufficient heat removal capability through the steam generators cause temperatures and pressures to decrease. When this capability is lost, at approximately 100 seconds, temperatures and pressures begin to increase.

The second and critical phase of the transient begins upon initiation of auxiliary feedwater at 915 seconds. The auxiliary feedwater flow capacity is sufficient to prevent primary coolant saturation and bulk boiling on the primary side (due to decay heat, and reactor coolant pump heat). The secondary heat sink remains effective in cooling the core and the core remains covered at all times.

The assumed auxiliary feedwater flow rate of 300 gpm is capable of removing decay heat and reactor coolant pump heat approximately 900 seconds after reactor trip.

The Reactor Coolant System and Main Steam System Pressures remain below 110% of the respective design pressures.

Based on the analysis results presented, the reactor core remains covered at all times, and therefore, in a coolable geometry. Since a reactor trip was assumed to occur on a secondary side trip signal only 5 seconds into the transient, well before the steam generator heat transfer capability is reduced, the primary system variables never approach a DNB condition.

The AFW system does not meet the single failure criterion (Ref. 3). The licensee in its submittal (Ref. 1) has stated that this concern will be addressed as part of the long term post-TMI requirement for the AFW system. (Task II.E.1 of the TMI Action Plan, NUREG-0660.)

VI CONCLUSION

As part of the SEP review for San Onofre 1, we have evaluated the licensee's analysis of the feedwater system pipe break against the criteria of SRP Section 15.2.8. The analysis has assumed the reactor coolant pumps to continue to run during the transient, thereby adding pump heat to the primary system, while the steam generators are blowing down. It is uncertain, however, whether the pump heat contribution during the transient results in a more conservative analysis than when the pumps are assumed tripped simultaneously with the event whereby the primary system will be in natural circulation. However, we believe that the effects on the primary system pressure and DNB would not be significant if natural circulation was assumed. Therefore, we find the analysis acceptable pending resolution of the single failure concern on the AFW system.