February 24, 1982

Docket No. 50-206
LS05-82-02-114

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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 1 - SEP TOPIC XV-9, STARTUP OF AN INACTIVE LOOP OR RECIRCULATION LOOP AT AN INCORRECT TEMPERATURE, AND FLOW CONTROLLER MALFUNCTION CAUSING AN INCREASE IN BWR CORE FLOW RATE

Enclosed is a copy of our final evaluation of SEP Topic XV-9 for San Onofre Unit 1. This evaluation incorporates comments from your letter of June 30, 1981 and hasbbeen rewritten in the standard SEP topic format.

This evaluation will be a basic input to the integrated safety assessment for your facility unless you identify changes needed to reflect the as-built conditions at your facility. This assessment may be revised in the future if your facility design is changed or if NRC criteria relating to this subject is modified before the integrated assessment is completed.

Sincerely,

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

Enclosure:
As stated

cc w/enclosure:
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SURNAME	EMcKenna:dk	GCwalina	WRussell	WPaulson	DCrutchfield	GLainas	
DATE	2/18/82	2/18/82	2/18/82	2/19/82	2/22/82	2/ /82	

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Vice President
Nuclear Engineering and Operations
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Post Office Box 800
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Sincerely,

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

WTR

OFFICE	DL:SEPB	DL:SEPB/SL	DL:SEPB/BC	DL:ORB#5/PM	DL:ORB#5/BC	DL:AD/SA	
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TOPIC XV-9 STARTUP OF AN INACTIVE LOOP

SAN ONOFRE 1

I. INTRODUCTION

Operation of a reactor at power with a reactor coolant loop out of service (i.e., the reactor coolant pump shutdown) results in a decrease in the coolant temperature in that loop. A subsequent restart of the idle reactor coolant pump without bringing that loop closer to system average temperature would then inject colder water mized with the flow from the active loops into the core. The resultant reactivity insertion could exceed that which the Reactor Control System is able to follow, leading to an increase in power and pressure and subsequent reduction in margin to departure from nucleate boiling (DNB).

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 transients 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) set forth the criteria for the design of water-cooled reactors.

GDC 10 "Reactor Design" requires that the core and associated cooling, 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 occurrences.

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 20 "Protection System Functions" requires that the protection system be designed to initiate automatically the operation of reactivity control systems to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences.

GDC 26 "Reactivity Control System Redundance and Capability" requires that the reactivity control system 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.

GDC 28 "Reactivity Limits" requires that the reactivity control system 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.

III. RELATED SAFETY TOPICS

Various other SEP topics evaluate features of the reactor protection system. The effects of single failures on safe shutdown capability are considered under Topic VII-3. Topic IV-1.A addresses plant operation with one reactor coolant pump out of service.

IV. REVIEW GUIDELINES

The review is conducted in accordance with SRP 15.4.4 and 15.4.5.

The evaluation included 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

San Onofre 1, a three loop plant, has no valves to isolate a reactor coolant loop; therefore, operation of two reactor coolant pumps will result in backflow through the idle loop, in turn causing a reverse Δ T in the loop, i.e., the temperature of the normal hot leg falls below that of the reactor inlet. A temperature interlock and alarm is provided for each loop to prevent starting a reactor coolant pump in an inactive loop if a reverse Δ T exists greater than a present value. Restarting an idle loop can be accomplished by reducing load which reduces the Δ T in the active loops to a value which will allow the idle loop pump to start. By Technical Specifications, the power level for operation with a loop out of service is limited to 10%. Furthermore, operation with only one or two reactor coolant pumps operating (power $\leq$ 10%) is limited to less than 24 hours.

Since there is no means of isolating a steam generator, the maximum idle loop temperature depression (with very small backflow) is limited by the temperature of the turbine cycle, thus providing an inherent limit on the severity of the transient. The range of possible inverse Δ T conditions is determined for a range of predicted backflow rates, and an analysis of the core response was performed by the licensee in the Final Safety Analysis for the most severe cases assuming the Δ T interlock to fail.

The transient following startup of an inactive loop is calculated with a detailed plant simulation by computer. A conservative calculation is made for low backflow and a conservative estimate of the overall heat transfer coefficient for the idle steam generator is made. These conditions yield the maximum idle loop temperature depression and the maximum initial power with two out of three loops. The reverse Δ T interlock and alarm will prevent a restart of the idle loop until the temperature difference is less than 5^{O}F. For this transient the interlock is assumed to be inoperative. The pump is assumed to accelerate to full flow instantaneously and the entire idle loop hot leg at the reduced temperature is swept into the core. A conservative maximum negative moderator coefficient is assumed. As a limiting condition the cold water is assumed to reach the core without mixing with the active loop flows, but is assumed to mix with the water in the inlet plenum. The core response is calculated assuming a temperature change is occurring throughout the core. A low value of Doppler coefficient is assumed thus yielding the maximum power increase as a result of the cold water.

For the purpose of this transient calculation, the ratio of power level to coolant flow through the core is assumed to be a constant, since this yields the minimum steam pressure, and the greatest possible temperature decrease in the inactive loop. This assumption also results in the smallest initial margin to DNB. Since total core flow is 65% (see below) and reactor power is limited to 10% by Technical Specification, this assumption is conservative. A high heat transfer coefficient from reactor cycle to turbine cycle in the inactive loop is used. This is approximately 85% of its normal full load value, whereas, the heat transfer coefficient will decrease a greater amount at the low load level at which the inactive loop is operating - approximately 18% of its normal full load value.

The licensee's results show that a backflow rate of 20,800 gpm in an idle loop was determined for operation with two out of three reactor coolant pumps running. The flow rate through the active loops is 74,500 gpm each, an increase over the normal flow with three loop operation. The resulting total core flow is then 128,200 gpm or about 65 percent of normal three loop flow. The resulting cold leg temperature in all loops is 542^{O}F and the hot leg temperature in the inactive loop is 505^{O}F. The steam generator temperature in the inactive loop is 500^{O}F so the inactive loop is approaching its minimum possible value using the high transfer coefficient. The hot leg temperature in the idle loop cannot go below the steam temperature. The assumed Doppler coefficient is -1×10^{-5} δk per ^{O}F and the moderator temperature coefficient is -3.5×10^{-4} δk per ^{O}F. The transient results shown in Figure 1 are computed assuming the full 37^{O}F temperature decrease is imposed at the inlet to the reactor vessel. Credit is taken for a mixing delay in the reactor inlet plenum but no credit for flow mixing with the two active loops.

The peak nuclear power of approximately 185%, based on Figure 1 is reached in about eight seconds. Thermal flux of about 85% is reached. The reactor trip is from overpower at 100 %. The licensee believes this is a conservatively high value since the overpower trip at this power level would be set to give a maximum trip level of 80% of full power.

Since the peak thermal power is less than full power and the inlet temperature is less than the full power value there is margin to DNB. The minimum DNB ratio is 3.1 and occurs at about 10 seconds.

Figure 2 shows the effect of flow mixing with the active loops. The temperature change at the reactor vessel inlet plenum then becomes -12^{o}F instead of -37^{o}F. All other parameters are identical to those for the transients shown in Figure 1. In the case of the reduction of flow mixing with the active loops the pressurizer's pressure peak is significantly reduced. The actual transient would lie somewhere between the two results and would most likely be represented by the results which assume flow mixing. The results also show that the change in pressurizer pressure is less than 100 psia.

VI. CONCLUSION

The San Onofre 1 plant has an interlock which prevents startup of an inactive loop unless the temperature in the idle loop is within 5^{o}F of the temperature in the other loops. Even under the assumption that this interlock fails, the consequences have been shown to satisfy the criteria. Therefore, we conclude that with respect to this topic, the San Onofre analysis meets current criteria and is acceptable.

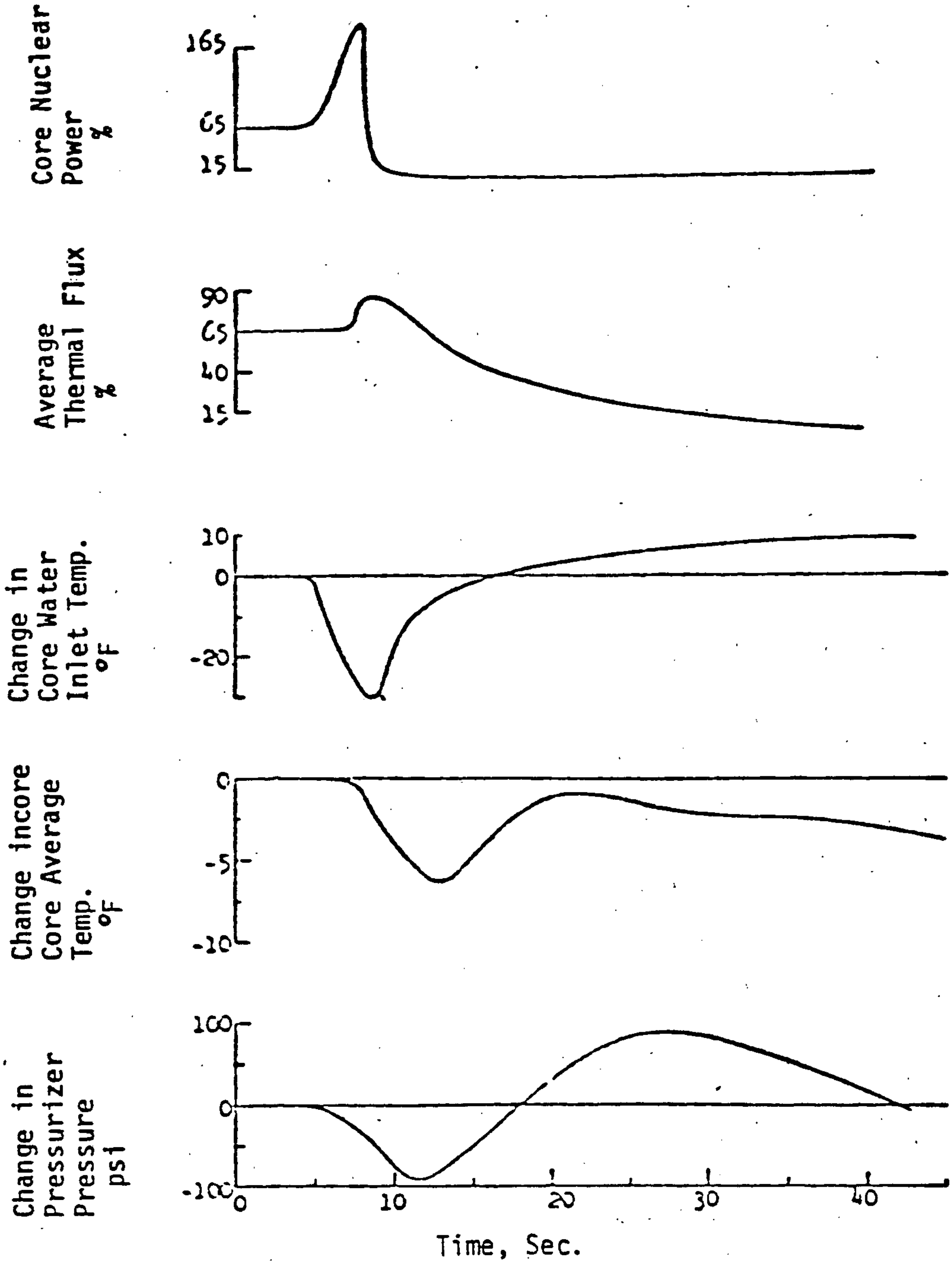

Figure 1
Transient Response for Startup of an Inactive Loop With Full Temperature Decrease

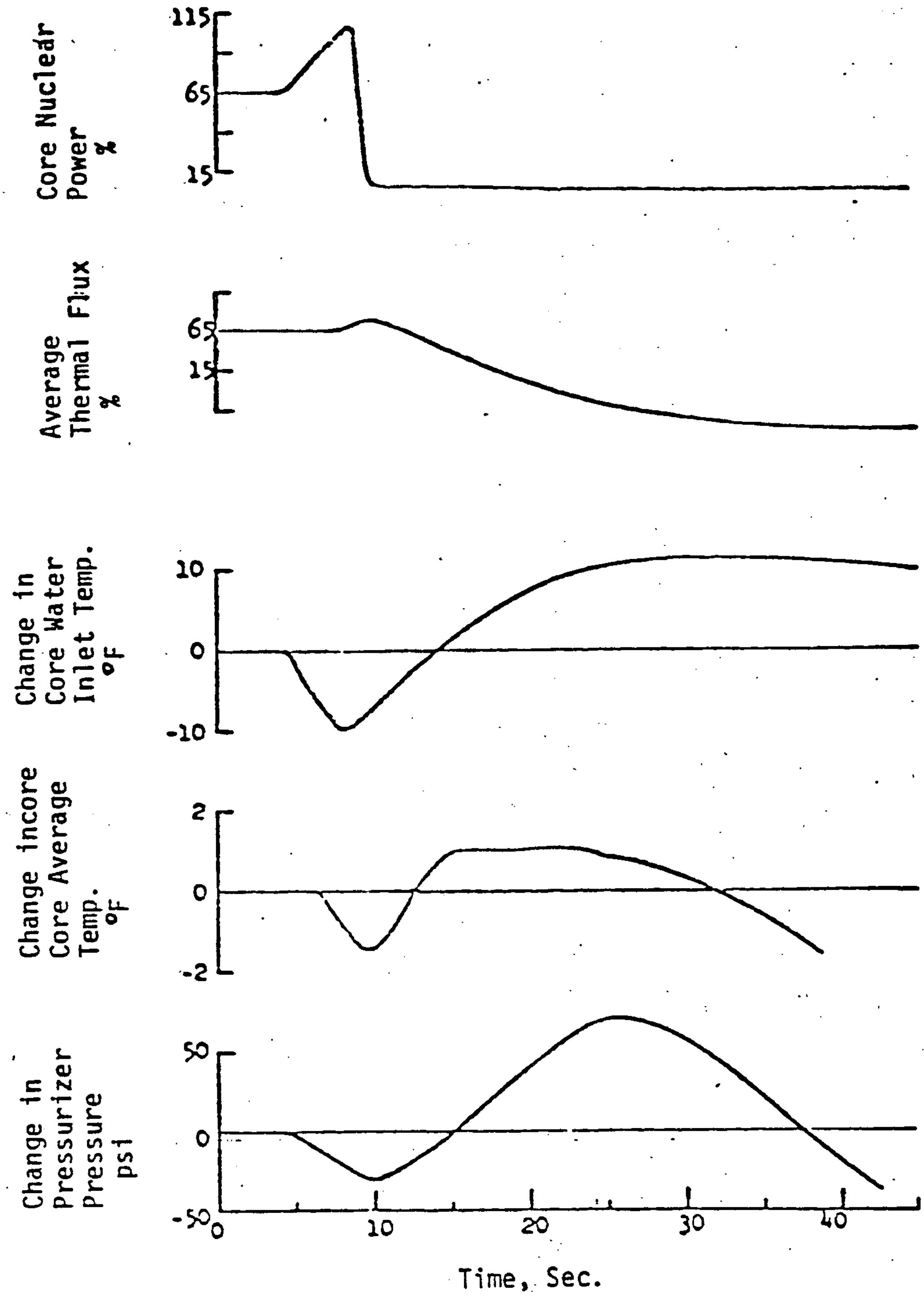

Figure 2
Transient Response for Startup of an Inactive Loop Without Full Temperature Decrease