SAN ONOFRE NUCLEAR GENERATING STATION UNIT 1

March 11, 1982

REACTOR COOLANT PUMP
ROTOR SEIZURE/SHAFT BREAK
ACCIDENT ANALYSIS

Work Conducted Under

Southern California Edison Company Purchase Order SOS00917

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the peak pressure. The thermal behavior of the fuel located at the core hot spot is investigated using the FACTRAN[2] code, using the core flow and the nuclear power calculated by LOFTRAN. The FACTRAN code includes the use of a film boiling heat transfer coefficient.

At the beginning of the postulated locked rotor accident (i.e., at the time the shaft in one of the RCPs is assumed to seize) the plant is assumed to be in operation under the most adverse steady-state operating conditions (i.e., maximum steady-state power level, maximum steady-state pressure, and maximum steady-state coolant average temperature). Plant characteristics and initial conditions are as assumed in most recent FSA analyses.

When the peak RCS pressure is evaluated, the initial pressure is conservatively estimated as 30 psi above nominal pressure (2100 psia) to allow for errors in the pressurizer pressure measurement and control channels. This is done to obtain the highest possible rise in the coolant pressure during the transient. To obtain the maximum pressure in the primary side, conservatively high loop pressure drops are added to the calculated pressurizer pressure. The pressure responses shown in figure 2 is the response at the point in the RCS having the maximum pressure.

Evaluation of the Pressure Transient

After pump seizure, the neutron flux is rapidly reduced by control rod insertion. Rod motion begins one second after the flow in the affected loop reaches 82 percent of nominal flow. No credit is taken for the pressure reducing effect of the pressurizer relief valves, pressurizer spray, steam dump or controlled feedwater flow after plant trip.

Although these operations are expected to occur and would result in a lower peak RCS pressure, an additional degree of conservatism is provided by ignoring their effect.

REACTOR COOLANT PUMP ROTOR SEIZURE (LOCKED ROTOR)

Identification of Causes and Accident Description

The accident postulated is an instantaneous seizure of an RCP rotor. Flow through the affected reactor coolant loop is rapidly reduced, leading to an initiation of a reactor trip on a low flow signal.

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generators is reduced, first because the reduced flow results in a decreased tube side film coefficient and then because the reactor coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators, causes an insurge into the pressurizer and a pressure increase throughout the RCS. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves, in that sequence. The power-operated relief valves are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure reducing effect as well as the pressure reducing effect of the spray is not included in the analysis.

Analysis of Effects and Consequences

Method of Analysis

Two digital computer codes are used to analyze this transient. The LOFTRAN[1] code is used to calculate the resulting loop and core flow transients following the pump seizure, the time of reactor trip based on the loop flow transients, the nuclear power following reactor trip, and to determine

The pressurizer safety valves are full open at 2575 psia and their capacity for steam relief is as described in most recent FSA analyses.

Evaluation of DNB in the Core During the Accident

For this accident, DNB is assumed to occur in the core, and therefore an evaluation of the consequences with respect to fuel rod thermal transients is performed. Results obtained from analysis of this "hot spot" condition represent the upper limit with respect to clad temperature.

In the evaluation, the rod power at the hot spot is assumed to be 3.0 times the average rod power (i.e., $F_Q = 3.0$) at the initial core power level.

Film Boiling Coefficient

The film boiling coefficient is calculated in the FACTRAN[2] code using the Bishop-Sandberg-Tong film boiling correlation. The fluid properties are evaluated at film temperature (average between wall and bulk temperatures). The program calculates the film coefficient at every time step based upon the actual heat transfer conditions at the time. The neutron flux, system pressure, bulk density, and mass flow rate as a function of time are used as program input.

For this analysis, the initial values of the pressure and the bulk density are used throughout the transient since they are the most conservative with respect to clad temperature response. For conservatism, DNB was assumed to start at the beginning of the accident.

Fuel Clad Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and clad (gap coefficient) has a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between pellet and clad. Based on investigations on the effect of the gap coefficient upon the maximum clad temperature during the transient, the

gap coefficient was assumed to increase from a steady-state value consistent with initial fuel temperature to 10,000 Btu/hr-ft^2-°F at the initiation of the transient. Thus, the large amount of energy stored in the fuel because of the small initial value is released to the clad at the initiation of the transient.

Results

The transient results for the most limiting of the locked rotor and pump shaft break accidents are shown in figures 1 through 4. The results of these calculations are also summarized in table 1. The peak RCS pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits. Also, the peak clad surface temperature is considerably less than 2700°F. It should be noted that the clad temperature was conservatively calculated assuming that DNB occurs at the initiation of the transient.

The calculated sequence of events for the cases analyzed is shown on table 1. Figure 1 shows that the core flow reaches a new equilibrium value by 10 seconds. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

Conclusions

A. Since the peak RCS pressure reached during any of the transients is less than that which would cause stresses to exceed the faulted condition stress limits, the integrity of the primary coolant system is not endangered.

B. Since the peak clad surface temperature calculated for the hot spot during the worst transient remains considerably less than 2700°F., the core will remain in place and intact with no loss of core cooling capability. No fuel failures are predicted for the locked rotor accident.

REACTOR COOLANT PUMP SHAFT BREAK

Identification of Causes and Accident Description

The accident is postulated as an instantaneous failure of an RCP shaft. Flow through the affected reactor coolant loop is rapidly reduced, though the initial rate of reduction of coolant flow is greater for the RCP rotor seizure event. Reactor trip is initiated on a low flow signal in the affected loop.

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generators is reduced, first because the reduced flow results in a decreased tube side film coefficient and then because the reactor coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators causes an insurge into the pressurizer and a pressure increase throughout the reactor coolant system. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves, in that sequence. The power-operated relief valves are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure reducing effect as well as the pressure reducing effect of the spray is not included in the analysis.

Conclusions

The consequences of an RCP shaft break are similar to those calculated for the locked rotor incident. With a failed shaft, the impeller could conceivably be free to spin in a reverse direction as opposed to being fixed in position as assumed in the locked rotor analysis. However, the net effect on core flow is negligible, resulting in only a slight decrease in the endpoint (steady-state) core flow. For both the shaft break and locked rotor incidents, reactor trip occurs very early in the transient. In addition, the analysis conservatively assumes that DNB occurs at the beginning of the transient.

The results shown in figures 1 through 4 and tables 1 and 2 represent the most limiting conditions for the locked rotor and pump shaft break accidents.

REFERENCES

1. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907, Revision 1, June, 1972.

2. Hargrove, H. B., "FACTRAN, a FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908, June, 1972.

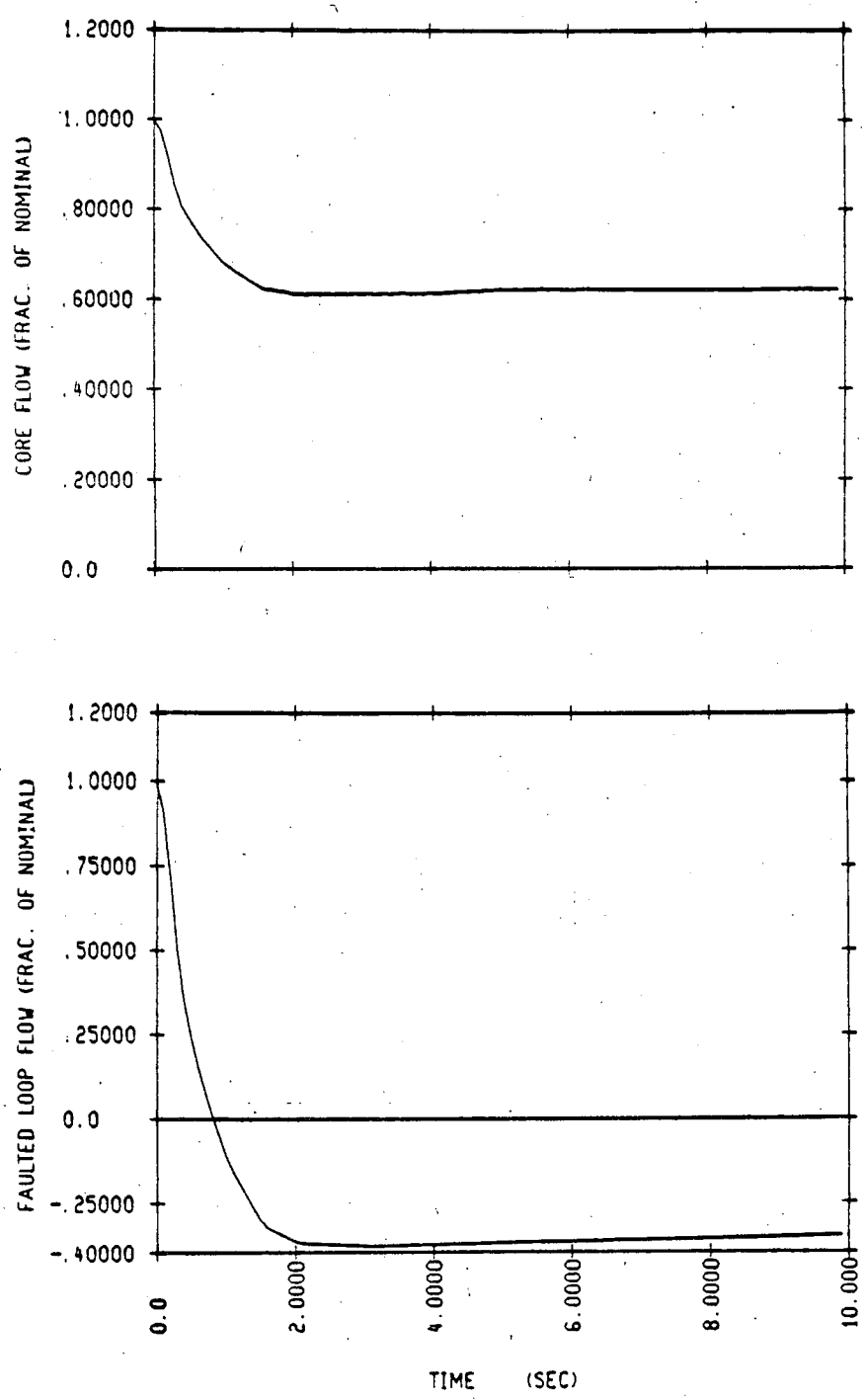

FIGURE 1

SAN ONOFRE, ONE LOCKED ROTOR

CORE AND FAULTED LOOP FLOW VS. TIME

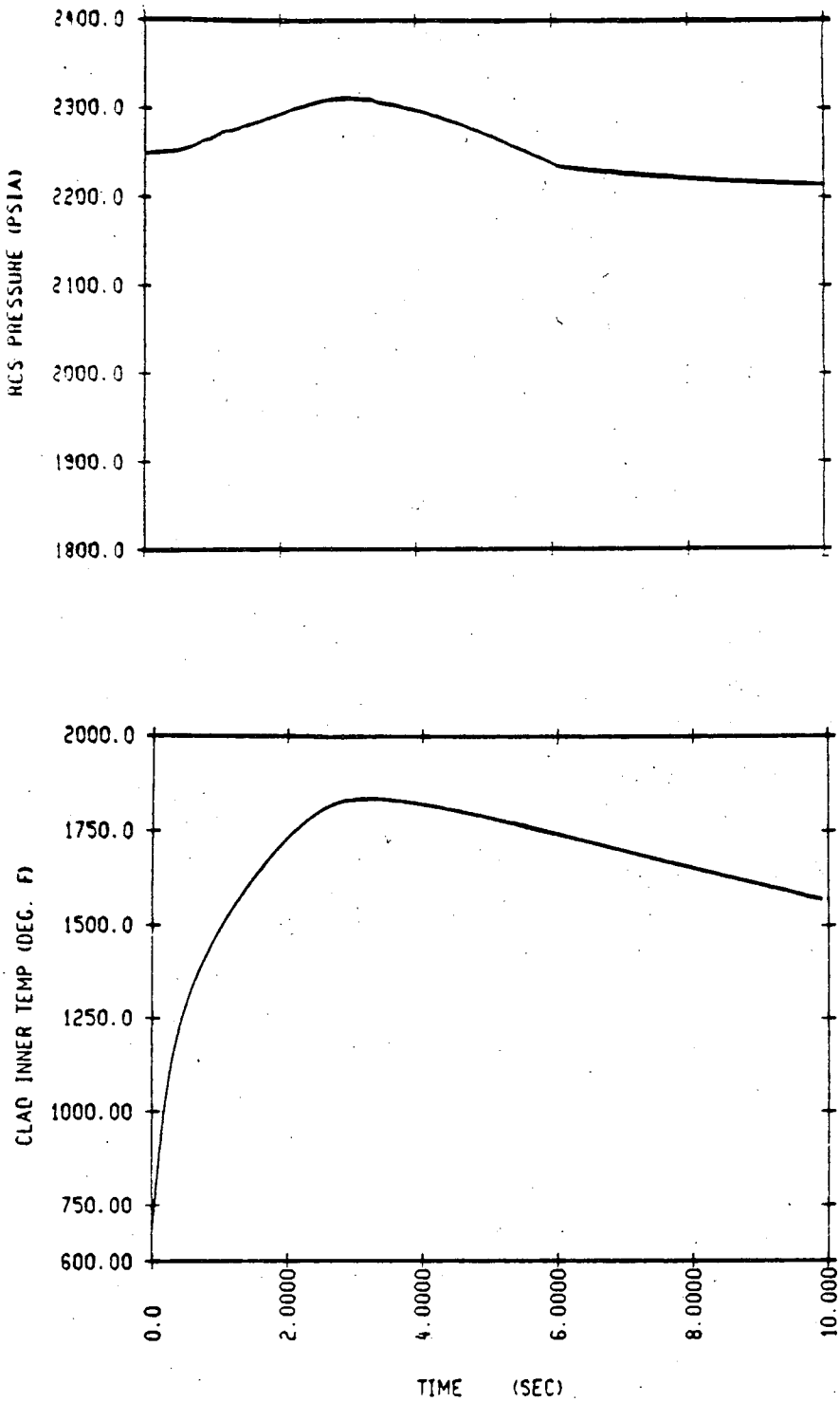

FIGURE 2

SAN ONOFRE, ONE LOCKED ROTOR
RCS PRESSURE AND CLAD TEMPERATURE
AT HOT SPOT VS. TIME

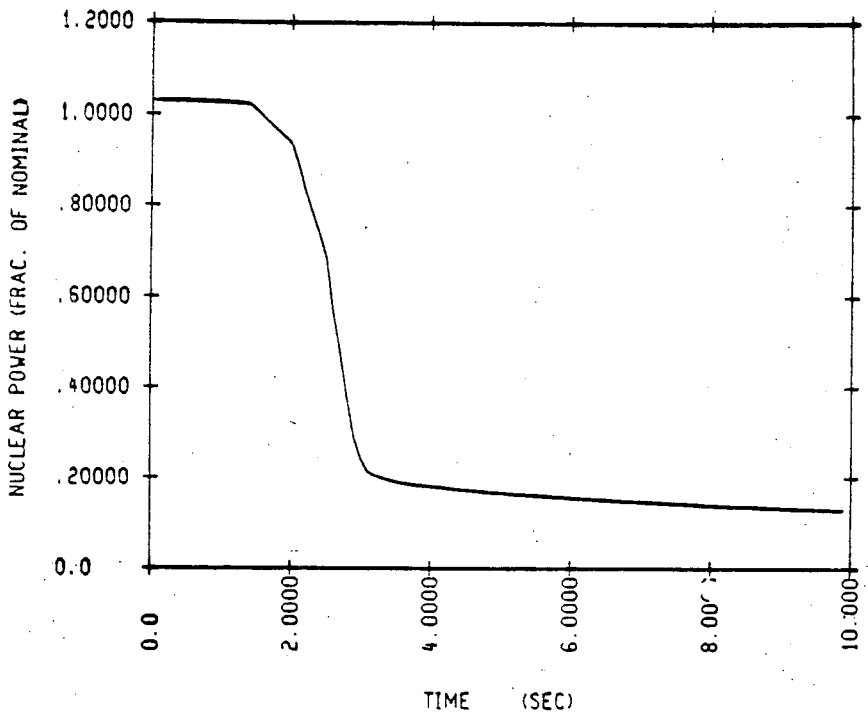

FIGURE 3

SAN ONOFRE, ONE LOCKED ROTOR
NUCLEAR POWER VS. TIME

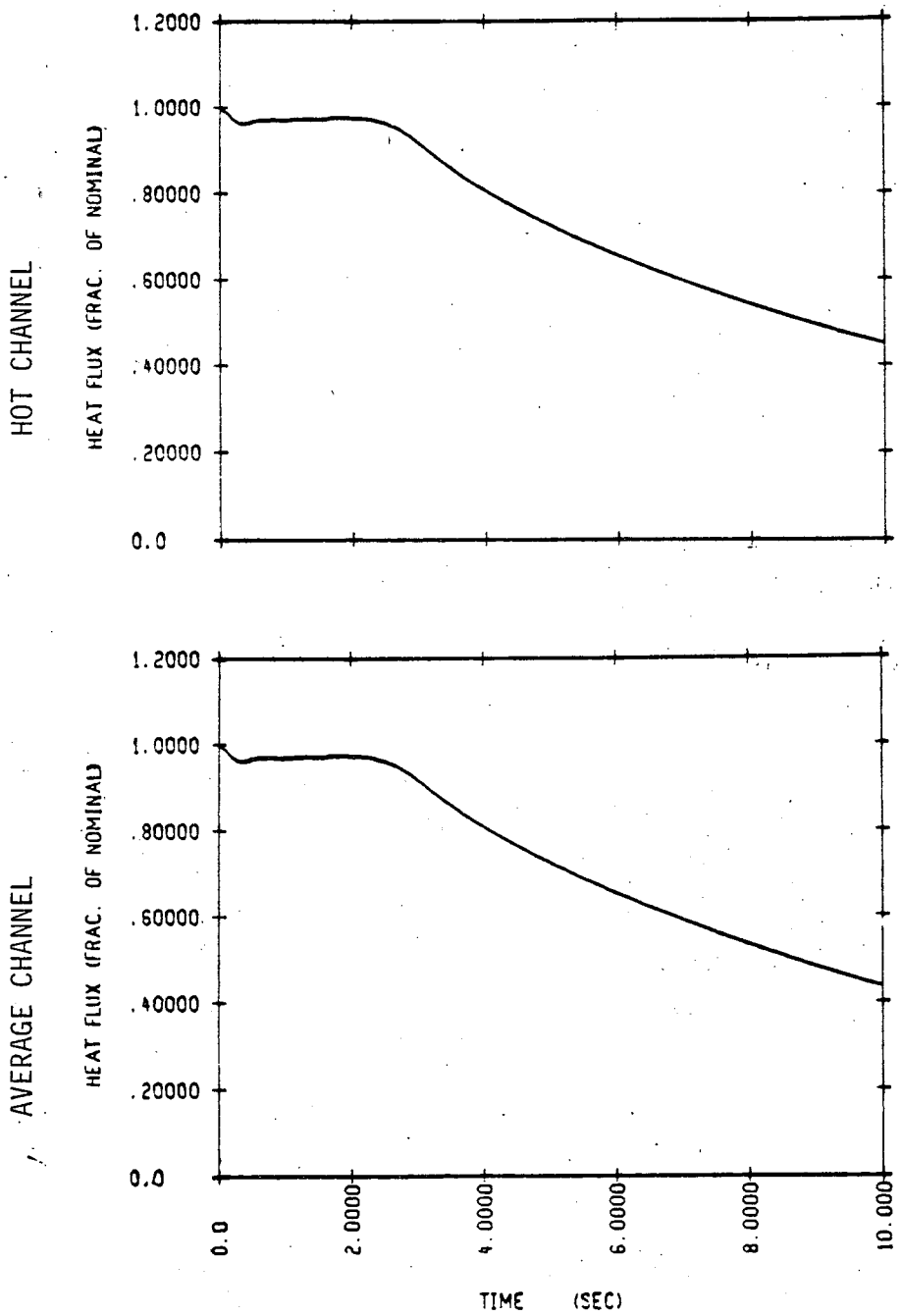

FIGURE 4

SAN ONOFRE, ONE LOCKED ROTOR

AVERAGE AND HOT CHANNEL HEAT FLUX VS. TIME

TABLE 1

TIME SEQUENCE OF EVENTS FOR REACTOR
COOLANT PUMP SHAFT SEIZURE TRANSIENT

Event	Time (sec)
Rotor in one pump locks	0.00
Low flow trip point reached	0.15
Rods begin to drop	1.35
Maximum RCS pressure occurs	2.9
Maximum clad temperature occurs	3.1

TABLE 2

SUMMARY OF RESULTS FOR REACTOR
COOLANT PUMP SHAFT SEIZURE TRANSIENT

Item	Maximum Value
Reactor coolant system pressure (psia)	2277
Clad temperature core hot spot (°F)	1835