EVALUATION OF REACTIVITY RESPONSE

FOR A STEAM LINE BREAK EVENT

WITH UNTERMINATED EMERGENCY

FEEDWATER FLOW

PREPARED FOR

FLORIDA POWER CORPORATION CR-3 PROJECT

CONTRACT #582-7087

TASK #184

MAY 6, 1981

B&W REFERENCE DOCUMENT: 86-1125549-00

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ABSTRACT

To reduce the potential for a loss-of-all-feedwater event, it has been recommended that the Steam Line Rupture Matrix Signal which isolates emergency feedwater be eliminated. A previous report, "Evaluation of Steam Line Break Consequences Associated with Removal of Rupture Matrix Signals from Emergency Feedwater Valves" addressed the effects of this change on the FSAR analysis, but did not specifically reanalyze the reactivity effects associated with a Steam Line Break (SLB). This report provides analyses, using current methods, which demonstrate that a return to criticality will not occur following a SLB and unterminated Emergency Feedwater (EFW) flow.

I. Background

Following the CR-3 transient in February, 1980, Florida Power Corporation convened a Nuclear Safety Task Force to identify potential changes which could significantly improve plant safety and reduce the potential for major transients. One of the principal recommendations of this task force (and NUREG-0667) was that the Steam Line Rupture Matrix System (SLRMS) signals which close the emergency feedwater isolation valves be eliminated. Such a change would significantly reduce the potential for a loss-of-all-feedwater event and this significantly improve the overall safety of the plant.

In support of this change, however, it was necessary to address the effects on the FSAR steam line break analysis which had been performed assuming automatic isolation of the EFW by the SLRMS. A report entitled, "Evaluation of Steam Line Break Consequences Associated with Removal of Rupture Matrix Signals from Emergency Feedwater Valves", dated May 23, 1980 and revised on June 27, 1980 was prepared for that purpose. Using the same methods as were originally used in the FSAR, the analysis described in that report demonstrated the acceptability of continued EFW flow from the standpoint of containment response. However, the method originally used for reactivity response to an SLB (the SECRUP analog code) was no longer in use or available. In lieu of the reanalysis, the report included an evaluation showing that the probability of a SLB accompanied by any stuck control rod could be conservatively estimated to be less than 2.75×10^{-7} per reactor year. However, the NRC staff rejected this evaluation, and, as a consequence, the proposed change has not been made.

Florida Power Corporation has subsequently instituted a major program to upgrade the emergency feedwater system and associated controls. The new control system, the Emergency Feedwater Initiation and Control (EFIC) system will replace the existing SLRMS and provide emergency feedwater isolation to only the steam generator in the affected loop. However, pending installation of the EFIC system, removal of the rupture matrix signals to the EFW valves is still a very desirable change.

This report is intended to address the NRC concern associated with reactivity effects following a SLB with continued EFW flow. For this analysis, current methods (specifically the TRAP code described in topical report BAW-10128) are used in lieu of the methods employed in the original FSAR analysis.

II. Scope

The purpose of this analysis is to document the core total reactivity response to a large double-ended steam line break with unterminated emergency feedwater flow to both steam generators. As a result of the proposed removal of the Steam Line Break Rupture Matrix signals from the emergency feedwater valves, it is possible to have EFW flow to both the affected and unaffected steam generator loops. This situation requires operator action to recognize and isolate EFW from the unaffected steam generator loop. This analysis predicts the system response, specifically core reactivity, for the SLB event where this operator action has not been performed. Core reactivity during the transient is determined by the insertion of control rods, the insertion of boron by the HPI system and the reactivity feedback due to fuel temperature and moderator density changes. Of special interest is the continued cooling provided by EFW. It is desired that core subcriticality be maintained throughout the transient.

III. Method

The Double-Ended Steam Line Break event was analyzed using the TRAP2 (version 6) digital computer code. Major assumptions and input parameters are provided in Table 1. The transient examined is a double-ended rupture of a single main steam line. The analysis assumptions were chosen to provide a conservative response with respect to core overcooling thus maximizing the core reactivity response and thus increasing the potential for recriticality. For this reason, a conservatively large secondary inventory, end of cycle kinetics parameters, and appropriate response delay times were used. No loss of offsite power was assumed. Main steam isolation is assumed to occur by closure of the main steam isolation valves rather than the turbine stop valves, thus allowing additional overcooling. Control rod reactivity insertion is assumed to provide sufficient reactivity to account for power deficit and only a 1% shutdown margin at HZP conditions. Shutdown reactivity provided by boron in the HPI system is assumed to be supplied by only 1 HPI pump. Positive reactivity feedback provided by the reactivity coefficients has been conservatively estimated as end of life values to maximize reactivity feedback response.

The double-ended rupture of a 22 inch (ID) main steam line results in a rapid increase in steam flow, secondary depressurization and an increase in heat transfer across the steam generator. Overcooling of the primary pressure results in a low RC pressure trip at 1.1 seconds into the transient. Reactor trip initiates turbine trip and TSV closure, but this function was ignored in the analyses to provide conservative overcooling. The secondary steam pressure continues to decrease actuating a Steam Line Rupture Matrix signal (600 psia SG pressure) at 4.7 seconds into the transient. This signal initiates closure of the Main Steam Isolation valves and main feedwater isolation. EFW is subsequently initiated as a result of the loss of all main feedwater. The steam generator in the affected loop continues to depressurize and boil dry at a rate faster than the steam generator in the unaffected loop. Primary system pressure decreases to the HPI actuation signal at 5.3 seconds, thus starting HPI pumps and the addition of borated water to the primary system.

Core reactivity becomes negative immediately following reactor trip providing at least a 1% subcritical margin. This margin is diminished by the positive reactivity feedback caused by the decrease in core average temperature due to the overcooling. Increased negative reactivity is provided by the introduction of borated water by the HPI system. This negative reactivity competes with the positive reactivity caused by prolonged cooling by EFW injection. The borated water supplied by one HPI pump provides sufficient negative reactivity to overcome continued EFW flow, thus eventually resulting in increasing subcritical margin. A minimum subcritical margin of .10% $\Delta k/k$ occurs 16 seconds into the transient followed by an increasing subcritical margin.

III. Results

The resulting reactivity response shown in Figure 2 indicates that the core will remain subcritical throughout the transient with a minimum subcritical margin of .10% $\Delta k/k$ occurring at 16 seconds. Reactor power and reactor coolant pressure are shown in Figure 1. A sequence of events is provided in Table 2 and summary of pertinent results in Table 3.

The analysis indicates that removal of the Steam Line Rupture Matrix signal to the EFW valves will not result in sufficient overcooling to cause core recriticality with credit being assumed for only one HPI pump. Operator action would normally be expected to occur to terminate EFW flow to the affected steam generator loop, thus allowing it to boil completely dry and terminate its contribution to the overcooling.

IV. Conclusions

Since subcriticality can be maintained even for the conservative assumptions presented in this analysis, it is concluded that removal of the SLBRM signals to the EFW valves does not result in an unacceptable core reactivity response.

TABLE 1

MAJOR ASSUMPTIONS AND INPUT PARAMETERS

A. Thermal Hydraulics

Parameter	Value
1. Power level (102%), MWt	2619
2. RC pump heat, MWt	18
3. Primary Flow rate, lbm/hr	1.3×10^8
4. Secondary flow rate, lbm/hr	1.06×10^7
5. SG outlet pressure psia	925
6. Steam generator inventory, lbm	46200 lbm
7. Initial pressurizer inventory ft^3	800
8. EFW flow rate	
affected loop gpm	880
unaffected loop gpm	520
EFW temperature, F	40

B. Kinetics Parameters

Parameter	Value
Doppler coefficient $\Delta k/k/F$	-1.3×10^{-5}
Moderator coefficient $\Delta k/k/F$	-3.0×10^{-5}
Boron Worth ppm/% $\Delta k/k$	108
Shutdown margin, % $\Delta k/k$	1.0

C. Trips and Setpoint Times

Parameter	Value
1. Low pressure RPS trip, psig	1800
2. HPI actuation setpoint, RC pressure psig	1500
3. SLB rupture matrix signal psig	600
a. MSIV closure, delay, s.	2.5
stroke, s.	5.0
b. Main feedwater isolation, s.	17
4. EFW actuation delay (sec.)	50

TABLE 2

STEAM LINE BREAK SEQUENCE OF EVENTS

Event	Time - (seconds)
Rupture occurs	0.0
Reactor trip on low primary pressure	1.1
Control rods begin to fall	1.5
Low steam generator pressure occurs	4.7
HPI actuation signal reached	5.3
MSIV's closed	12.2
HPI flow established	15.3
Minimum subcritical margin reached	16.
Main feedwater isolated	24.2
EFW flow established	50.

TABLE 3

STEAM LINE BREAK RESULTS

Minimum subcritical margin, %Δk/k	.10
Time of minimum subcritical margin, s.	16

Figure 1A-TOTAL REACTOR POWER (% OF 2619 MWt) VS TRANSIENT TIME (SECONDS)

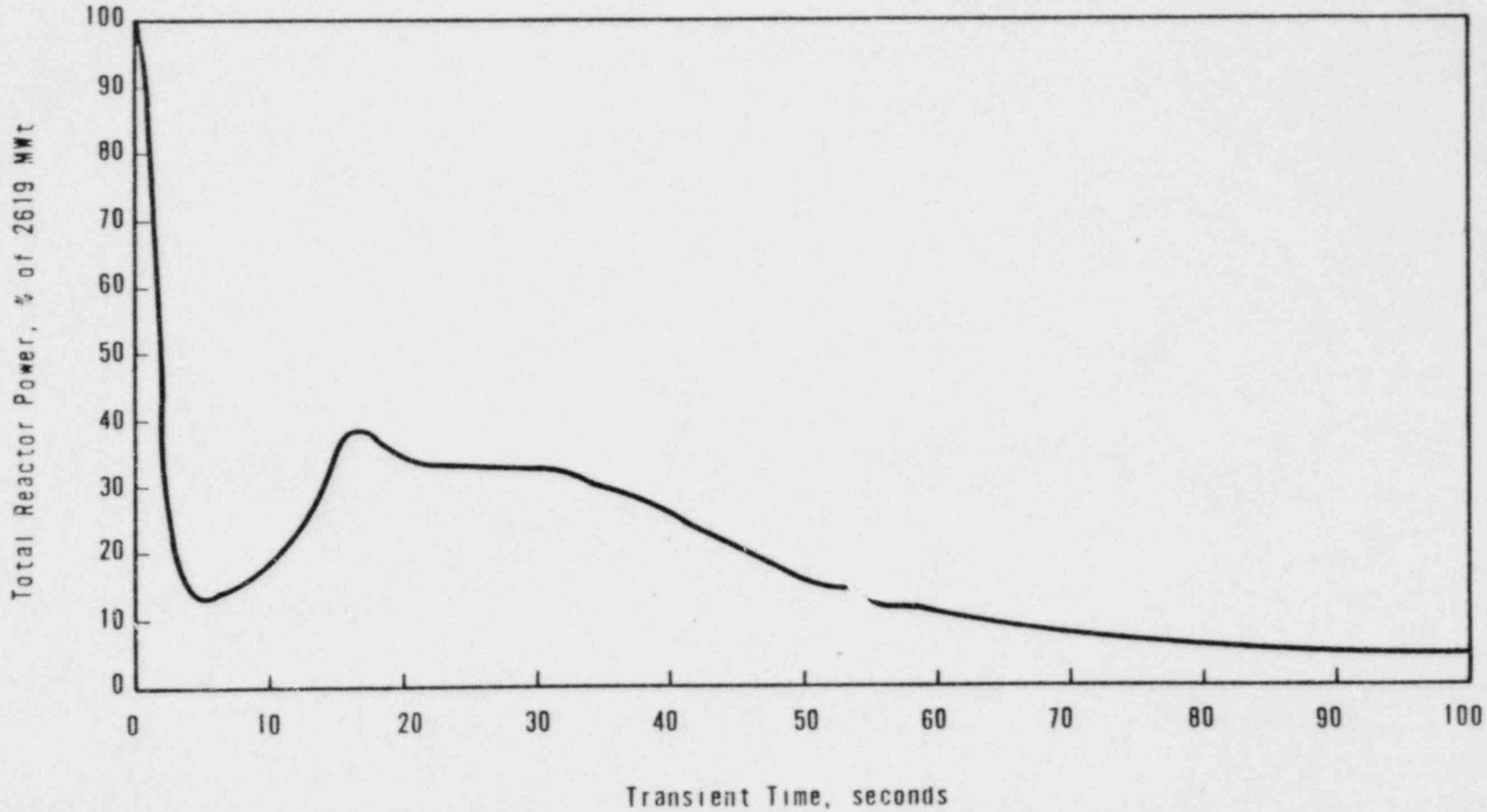

Figure 1B-AVERAGE REACTOR COOLANT TEMPERATURE (°F) VS TRANSIENT TIME (SECONDS)

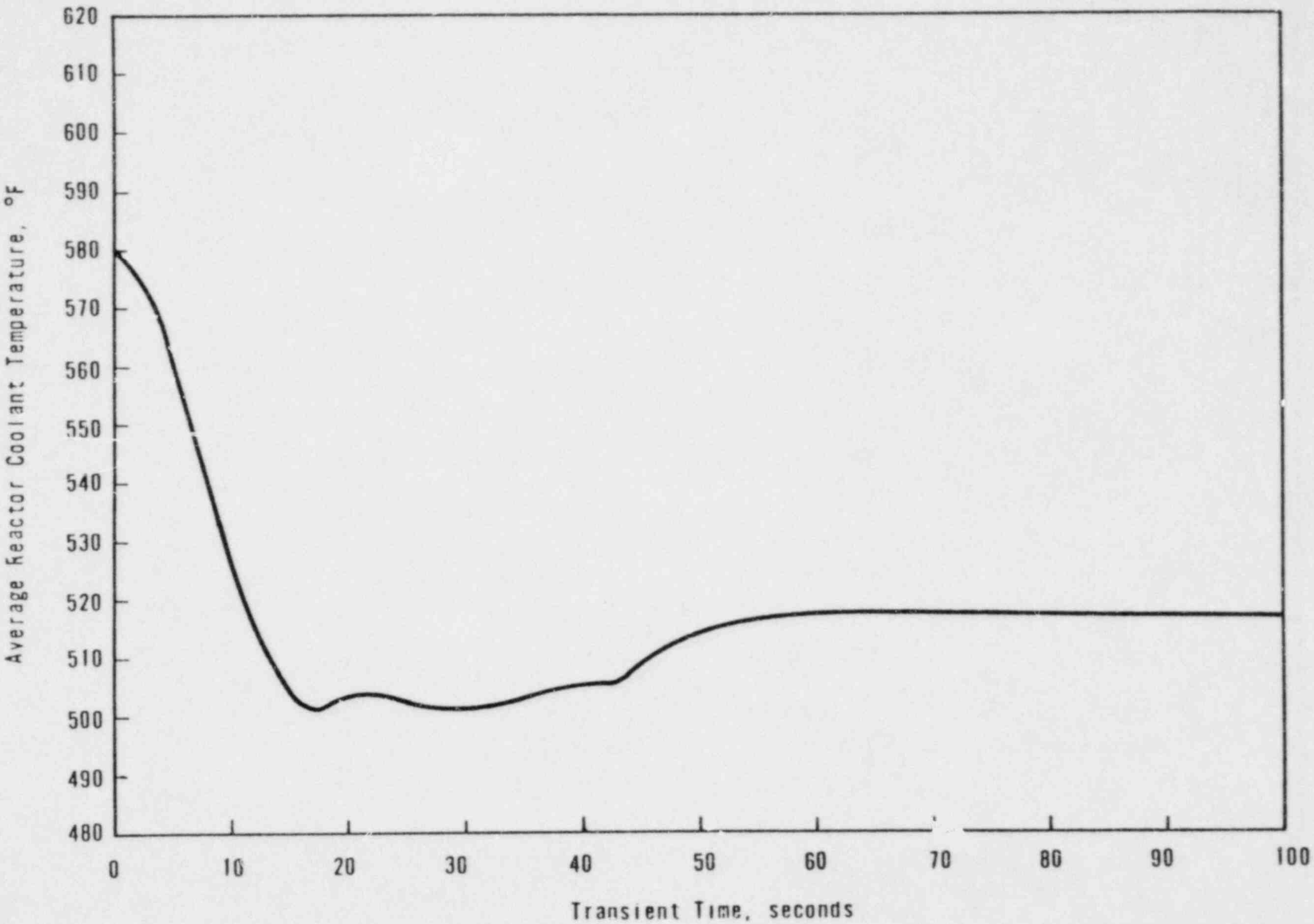

Figure 2-TOTAL REACTIVITY (% ΔK/K) VS TRANSIENT TIME (SECONDS)