14 SAFETY ANALYSIS

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14 SAFETY ANALYSIS

14.1 CORE AND COOLANT BOUNDARY PROTECTION ANALYSIS

14.1.1 ABNORMALITIES

In previous sections of this report both normal and abnormal operations of the various systems and components have been discussed. This section summarizes and further explores abnormalities that are either inherently terminated or require the normal protective systems to operate to maintain integrity of the fuel and/or the reactor coolant system. These abnormalities have been evaluated for rated power of 2,452 MWt. Whenever a fission product release to the environment occurs, the release is based upon the fission product inventory associated with the ultimate reactor core power level of 2,544 MWt. Fission product dispersion in the atmosphere is assumed to occur as predicted by the dispersion models developed in 2.3. Table 14-1 summarizes the potential abnormalities studied.

Table 14-1
Abnormalities Affecting Core and Coolant Boundary

Event	Cause	Effect
Uncompensated Operating Reactivity Changes	Fuel depletion or xenon buildup.	Reduction in reactor system average temperature. Automatic reactor trip if uncompensated. No equipment damage or radiological hazard.
Startup Accident	Uncontrolled rod(*) withdrawal.	Power rise terminated by negative Doppler effect, reactor trip from short period, high reactor coolant system pressure, or overpower. No equipment damage or radiological hazard.
Rod Withdrawal Accident at Rated Power	Uncontrolled rod withdrawal.	Power rise terminated by overpower trip or high pressure trip. No equipment damage or radiological hazard.

(*) Control rod, rod, and control rod assembly (CRA) are used interchangeably in this section and elsewhere in the report.

A control rod group consists of a symmetrical arrangement of four or more control rod assemblies. See 7.2.2.1.1.

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Table 14-1 (Cont'd)

Event	Cause	Effect
Moderator Dilution Accident	Equipment malfunction or operator error.	Slow change of power terminated by reactor trip on high temperature or pressure. During shutdown a decrease in shutdown margin occurs, but criticality does not occur. No radiological hazard.
Loss of Coolant Flow	Mechanical or electrical failure of reactor coolant pump(s).	None. Core protected by reactor low-flow trip. No radiological hazard.
Stuck-Out or Stuck-In or Dropped-In Control Rod	Mechanical or electrical failure.	None. Subcriticality can be achieved if one rod is stuck-out. If stuck-in or dropped-in, continued operation is permitted if effect on power peaking not severe. No radiological hazard.
Loss of Electric Power	Miscellaneous faults.	Possible power reduction or reactor trip depending on condition. Redundancy provided for safe shutdown. Radiological hazard within limits of 10 CFR 20.
Steam Line Failure	Pipe failure.	Reactor automatically trips if rupture is large. No damage to reactor system. Integrated doses at exclusion distance are 0.002 rem whole body and 0.53 rem thyroid. Radiological hazard is within limits of 10 CFR 20.
Steam Generator Tube Failures	Tube failure.	Reactor automatically trips if leakage exceeds normal makeup capacity to reactor coolant system. Integrated doses at exclusion distance are 0.69 rem whole body and 1.0 x 10^{-4} rem thyroid. Radiological hazard is within limits of 10 CFR 20.

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14.1.2 ANALYSIS OF EFFECTS AND CONSEQUENCES

14.1.2.1 Uncompensated Operating Reactivity Changes

14.1.2.1.1 Identification of Cause

During normal operation of the reactor, the overall reactivity of the core changes because of fuel depletion and changes in fission product poison concentration. These reactivity changes, if left either uncompensated or overcompensated, can cause operating limits to be exceeded. In all cases, however, the reactor protective system prevents safety limits from being exceeded. No damage occurs from these conditions.

14.1.2.1.2 Analysis and Results

During normal operation, the automatic reactor control system senses any reactivity change in the reactor. Depending on the direction of the reactivity change, the reactor power increases or decreases. Correspondingly, the reactor coolant system average temperature increases or decreases, and the automatic reactor control system acts to restore reactor power to the power demand level and to reestablish this temperature at its set point. If manual corrective action is not taken or if the automatic control system malfunctions, the reactor coolant system average temperature changes to compensate for the reactivity disturbance. Table 14-2 summarizes these disturbances.

Table 14-2
Uncompensated Reactivity Disturbances

Cause	Maximum Reactivity Rate, $(\Delta k/k)$/sec	Rate of Average Temperature Change (Uncorrected), F/sec
Fuel Depletion	-6×10^{-9}	-0.0006
Xenon Buildup	-3×10^{-8}	-0.003

These results are based on $+6 \times 10^{-5}$ $(\Delta k/k)$/F moderator coefficient and -1.14×10^{-5} $(\Delta k/k)$/F Doppler coefficient. The nominal value of $+6 \times 10^{-5}$ $(\Delta k/k)$/F is representative of the moderator coefficient at the beginning of core life for an equilibrium cycle. This value is also valid at BOL for the first cycle after 15 days. A higher value [$+10 \times 10^{-5}$ $(\Delta k/k)$/F] exists at the start of the first core cycle. However, the effect of this slightly higher value has been shown to be of minor importance by the evaluation of the sensitivity of the reactor to moderator coefficient variations. These reactivity changes are extremely slow and allow the operator to detect and compensate for the change. 1

14.1.2.2 Startup Accident

14.1.2.2.1 Identification of Cause

The objective of a normal startup is to bring a subcritical reactor to the critical or slightly supercritical condition, and then to increase power in a controlled manner until the desired power level and system operating temperatures are obtained. During a startup, an uncontrolled reactivity addition could cause a nuclear excursion. This excursion is terminated by the strong negative Doppler effect if no other protective action operates.

The following design provisions minimize the possibility of inadvertent continuous rod withdrawal and limit the potential power excursion:

a. The control system is designed so that only one control rod group can be withdrawn at a time, except that there is a 25 per cent overlap in travel between two successive rod groups. This overlap occurs at the minimum worth for each group since one group is at the end of travel and the other is at the beginning of travel. The maximum worth of any single control rod group is 1.2% $\Delta k/k$ when the reactor is critical. 7

b. Control rod withdrawal rate is limited to 30 in./min. 7

c. A short-period withdrawal stop and alarm are provided in the source range.

d. A short-period withdrawal stop, alarm, and trip are provided in the intermediate range.

e. A high flux level and a high pressure trip are provided in the power range.

The reactor protective system is designed to limit (a) the reactor thermal power to 114 per cent of rated power to prevent fuel damage, and (b) the reactor coolant system pressure to 2,515 psia.

14.1.2.2.2 Methods of Analysis

An analog model of the reactor core and coolant system was used to determine the characteristics of this accident. This analog model used full reactor coolant flow, but no heat transfer out of the system and no sprays in the pressurizer. The rated-power Doppler coefficient [-1.14×10^{-5} $(\Delta k/k)/F$] was used although the Doppler is much larger than this for the principal part of the transient. The rods were assumed to be moving along the steepest part of the rod-worth vs rod-travel curve. A reactor trip on short period was not incorporated in the analysis. The nominal values of the principal parameters used were: 0.3 sec trip delay, $+6 \times 10^{-5}$ $(\Delta k/k)/F$ moderator coefficient, and -1.14×10^{-5} $(\Delta k/k)/F$ Doppler coefficient. The total worth of all the control rods inserted into the reactor core following any trip is 8.4% $\Delta k/k$ without a stuck 1

Control rod, or 5.4% $\Delta k/k$ (the nominal case in this study) with a stuck rod. 1

14.1.2.2.3 Results of Analysis

Figure 14-1 shows the results of withdrawing the maximum worth control rod group from 1 per cent subcritical. This group is worth a maximum value of 1.2% $\Delta k/k$ and results in a reactivity addition rate of 5.8×10^{-5} 7
$(\Delta k/k)$/sec. The Doppler effect begins to slow the neutron power (*) rise, but the heat to the coolant increases the pressure past the trip point, and the transient is terminated by the high pressure trip.

Figure 14-2 shows the results of withdrawing all 61 control rod assemblies (with a total worth of 10.0% $\Delta k/k$) at 1 per cent subcritical. This results in a reactivity addition rate of 5.8×10^{-4} $(\Delta k/k)$/sec. About 15.3 sec after passing through criticality, the neutron power peaks at 147 per cent, where the power rise is stopped by the negative Doppler effect. The high neutron flux trip takes effect 0.25 sec after the peak power is reached and terminates the transient. The peak thermal heat flux is only 16 per cent of the rated power heat flux. 7

A sensitivity analysis was performed on both of these startup accidents to determine the effect of varying several key parameters. Figures 14-3 through 14-6 show typical results for the single group, 1.2% $\Delta k/k$ start-up accident.

Figures 14-3 and 14-4 show the effect of varying the reactivity addition rate on the peack thermal power and peak neutron power. This reactivity rate was varied from one order of magnitudy below the single rod group case (1.2% $\Delta k/k$) to more than an order of magnitude above the rate that 7
represents all rods (10.0% $\Delta k/k$) being withdrawn at once. The slower 1
rates - up to about 0.5×10^{-3} $(\Delta k/k)$/sec - will result in the pressure trip being actuated, whereas only the very fast rates actuate the high neutron flux level trip.

Figures 14-5 and 14-6 show the peak thermal power variation as a function of a wide range of trip delay times and Doppler coefficients for the 1.2% $\Delta k/k$ rod group. Only a small change in power is noted. Figures 14-7 and 14-8 are the corresponding results from the withdrawal of all rods
(10.0% $\Delta k/k$). Since this transient inserts reactivity an order of magni- 1
tude faster than does the single control rod group case, there is considerably more variation in the peak thermal power over these wide ranges. At high values of the Doppler coefficient, the neutron power rise if virtually stopped before reaching the high flux trip level. Reactor power generation continues until sufficient energy is transferred to the reactor coolant to initiate a high pressure trip. This results in a higher peak thermal power.

Figures 14-9 through 14-12 show the peak pressure response to variations in several key parameters for the case where all rods are withdrawn. It is seen that the safety valve is opened when these parameters are changed considerably from the nominal values, except in the case of the moderator

(*) Neutron power is defined as the total sensible energy release from fission.

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coefficient which has little effect because of the short duration of the transient. Again for a high Doppler coefficient, the high pressure trip is relied upon.

None of these postulated startup accidents, except for reactivity addition rates greater than 2×10^{-3} $(\Delta k/k)$/sec, which is three times greater than for withdrawal of all rods at once, causes a thermal power peak in excess of 40 per cent rated power or a nominal fuel rod average temperature greater than 1,715 F. The nominal 1.2% $\Delta k/k$ rod group withdrawal causes a peak pressure of 2,515 psia, the safety valve set point. The capacity of the safety valves is adequate to handle the maximum rate of coolant expansion resulting from this startup accident. The 10.0% 1
$\Delta k/k$ withdrawal - using all 69 rods - causes a peak pressure of only 2,465 psia because the flux trip is actuated prior to the pressure trip.

It is concluded that the reactor is completely protected against any startup accident involving the withdrawal of any or all control rods, since in no case does the thermal power approach 114 per cent, and the peak pressure never exceeds 2,515 psia.

14.1.2.3 <u>Rod Withdrawal Accident From Rated Power Operation</u>

14.1.2.3.1 Identification of Cause

A rod withdrawal accident presupposes an operator error or equipment failure which results in accidental withdrawal of a control rod group while the reactor is at rated power. As a result of this assumed accident, the power level increases; the reactor coolant and fuel rod temperatures increase; and if the withdrawal is not terminated by the operator or protection system, core damage would eventually occur.

The following provisions are made in the design to indicate and terminate this accident:

a. High reactor outlet coolant temperature alarms.

b. High reactor coolant system pressure alarms.

c. High pressurizer level alarms. 1

d. High reactor outlet coolant temperature trip.

e. High reactor coolant system pressure trip.

f. High power level trip.

14.1.2.3.2 Methods of Analysis

An analog computer model was used to determine the characteristics of this accident. A complete kinetics model, pressure model, average fuel rod model, steam demand model with turbine coastdown to 15 per cent of rated load, coolant transport model, and a simulation of the instrumentation for pressure and flux trip were included. The initial conditions were normal rated power operation without automatic control. Only the

moderator and Doppler coefficient of reactivity were used as feedback. The nominal values used for the main parameters were 0.3 sec trip delay time, -1.14×10^{-5} ($\Delta k/k$)/F Doppler coefficient, $+6 \times 10^{-5}$ ($\Delta k/k$)/F moderator coefficient, 5.8×10^{-5} ($\Delta k/k$)/sec reactivity insertion rate, and 1.2% $\Delta k/k$ 7
control rod group worth. The total worth in all the control rods inserted into the reactor core following any trip is 8.4% $\Delta k/k$ without a stuck control 1
rod, or 5.4% $\Delta k/k$ (the nominal value used) with a stuck rod.

(Sentence deleted.) 7

The reactor protection system is designed to limit (a) the reactor power to 114 per cent of rated power to prevent fuel damage, and (b) the coolant system pressure to 2,515 psia to prevent reactor coolant system damage.

14.1.2.3.3 Results of Analysis

Figure 14-13 shows the results of the nominal rod withdrawal from rated power using the 1.2% $\Delta k/k$ rod group at 5.8×10^{-5} ($\Delta k/k$)/sec. The transient is terminated by a high pressure trip, and reactor power is limited to 108 per cent, much less than the design overpower of 114 per cent of rated power. The changes in the parameters are all quite small, e.g., 5 F average reactor coolant temperature rise and 200 psi system pressure change.

A sensitivity analysis of important parameters was performed around this nominal case, and the resultant reactor coolant system pressure responses are shown in Figures 14-14 through 14-16.

Figure 14-14 shows the pressure variation for a very wide range of rod withdrawal rates - more than an order of magnitude smaller and greater than the nominal case. For the very rapid rates, the neutron flux level trip is actuated. This is the primary protective device for the reactor core; it also protects the system against high pressure during fast rod withdrawal accidents. The high pressure trip is relied upon for the slower transients. In no case does t e thermal power exceed 108 per cent rated power.

An analysis has been performed extending the evaluation of the rod withdrawal accident for various fractional initial power levels up to rated power. This evaluation has been performed assuming simulated withdrawal of all 61 control rods 7
with a reactivity addition rate of 5.8×10^{-4} ($\Delta k/k$)/sec. This rate is a factor of ten higher than used in the cases evaluated at rated power. The results of this analysis are shown in Figures 14-17 and 14-18.

As seen in Figure 14-17 the peak thermal power occurs for the rated power case and is well below the maximum design power of 114 per cent. The peak neutron power for all cases is approximately 117 per cent of rated power and represents a slight overshoot above the trip level of 114 per cent. Figure 14-18 shows that the maximum fuel temperature reached in the average rod and the hot spot are well below melting. Even in the most severe case at rated power, the average fuel temperature only increases by 26 F. It is therefore readily concluded that no fuel damage would result from simultaneous all-rod withdrawal from any initial power level.

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Figures 14-15 and 14-16 show the pressure response to variations in the trip delay time and Doppler coefficient. For the higher values of the Doppler coefficient, the pressure trip is always actuated, and, therefore, the pressure levels off.

This analysis shows that the high pressure trip and the high flux level trip adequately protect the reactor against any rod withdrawal accident from rated power.

14.1.2.4 Moderator Dilution Accident

14.1.2.4.1 Identification of Cause

The reactor utilizes boric acid in the reactor coolant to control excess reactivity. The boron content of the reactor coolant is periodically reduced to compensate for fuel burnup. The dilution water is supplied to the reactor coolant system by the makeup and purification system. This system is designed with several interlocks and alarms to prevent improper operation. These are as follows:

a. Flow of dilution water to the makeup tank must be initiated by the operator. The dilution water addition valve can be opened only when the control rods have been withdrawn to the preset position (95 per cent) and the timing device to limit the integrated flow has been set. Dilution water is added at flow rates up to 70 gpm.

b. Flow of dilution water is automatically stopped when either the flow has integrated to a preset value or when the rods have been inserted to a preset position (at about 75 per cent full stroke).

c. A warning light is on whenever dilution is in progress.

The makeup and purification system normally has one pump in operation which supplies up to 70 gpm to the reactor coolant system and the required flow to the reactor coolant pump seals. Thus, the total makeup flow available is limited to 70 gpm unless the operator takes action to increase the amount of makeup flow to the reactor coolant system. When the makeup rate is greater than the maximum letdown rate of 70 gpm, the net water makeup will cause the pressurizer level control to close the makeup valves. 2

The nominal moderator dilution event considered is the pumping of water with zero boron concentration from the makeup tank to the reactor coolant system by the makeup pump.

It is also possible, however, to have a slightly higher flow rate during transients when the system pressure is lower than the nominal value and the pressurizer level is below normal. This flow might be as high as 100 gpm.

In addition, with a combination of multiple valve failures or maloperations, plus more than one makeup pump operating and reduced reactor coolant system pressure, the resulting inflow rate can be as high as 500 gpm. This constitutes the maximum dilution accident. A reactor trip would terminate unborated water addition to the makeup tank, and total flow into the coolant system would be terminated by a high pressurizer level.

The criteria of reactor protection for this accident are

a. The reactor power will be limited to less than the design overpower of 114 per cent rated power to prevent fuel damage.

b. The reactor protection system will limit the reactor coolant system pressure to less than the system design pressure of 2,500 psig.

c. The reactor minimum subcriticality margin of 1% $\Delta k/k$ will be maintained.

d. Administrative procedures will be imposed to monitor and control the relationship of control rod regulating group patterns and boron concentrations in the reactor coolant over the operating life of the core. 1

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14.1.2.4.2 Analysis and Results

The reactor is assumed to be operating at rated power with an initial boron concentration (1,800 ppm) in the reactor coolant system. The dilution water is uniformly distributed throughout the reactor coolant volume. Uniform dilution results from a discharge rate of 70-500 gpm into a reactor coolant flow of 88,000 gpm. A change [illegible] concentration of 100 ppm produces a 1% $\Delta k/k$ reactivity change. The effects of these three dilution rates on the reactor are as follows:

Dilution Water Flow, gpm	Reactivity Rate, $(\Delta k/k)$/sec	Average Reactor Coolant System Temp. Change, F/sec
70	$+2.5 \times 10^{-6}$	0.3
100	$+3.6 \times 10^{-6}$	0.3
500	$+1.8 \times 10^{-5}$	0.4

The fastest rate of dilution can be handled by the automatic control system, which would insert rods to maintain the power level and reactor coolant system temperature. If an interlock failure occurred while the reactor was under manual control, these reactivity additions would cause a high reactor coolant temperature trip or a high pressure trip. In any event the thermal power will not exceed 114 per cent rated power, and the system pressure will not exceed the design pressure of 2,500 psig. Therefore moderator dilution accidents will not cause any damage to the reactor system.

During refueling or maintenance operations when the reactor closure head has been removed, the sources of dilution water makeup to the makeup tank--and therefore to the reactor coolant system--are locked closed, and the makeup pumps are not operating. At the beginning of core life when the boron concentration is highest, the reactor is about 9.5% $\Delta k/k$ subcritical with the maximum worth rod stuck out. To demonstrate the ability of the reactor to accept moderator dilution during shutdown, the consequences of accidentally filling the makeup tank with dilution water and starting the makeup pumps have been evaluated. The entire water volume from the makeup tank could be pumped into the reactor coolant system (assuming only the coolant in the reactor vessel is diluted), and the reactor would still be 6.5% $\Delta k/k$ subcritical.

14.1.2.5 Cold Water Accident

The absence of individual loop isolation valves eliminates the potential source of cold water in the reactor coolant system. Therefore, this accident is not credible in this reactor.

14.1.2.6 Loss-of-Coolant Flow

14.1.2.6.1 Identification of Cause

A reduction in the reactor coolant flow rate occurs if one or more of the reactor coolant pumps should fail. A pumping failure can occur from mechanical failures or from a loss of electrical power. With four independent pumps available, a mechanical failure in one pump will not affect operation of the others.

Each reactor coolant pump receives electrical power from one of the two electrically separate busses of the 6,900 volt system discussed in 8.2.2.3. Loss of a unit auxiliary transformer to which the 6,900 volt busses are normally connected will initiate a rapid transfer to the startup transformer source without loss of coolant flow. Faults in an individual pump motor or its power supply could cause a reduction in flow, but a complete loss of flow is extremely unlikely.

In spite of the low probability of a complete loss of power to all reactor coolant pumps, the nuclear unit has been designed so that such a failure would not lead to core damage.

The reactor protection criterion for loss-of-coolant flow conditions starting at rated power is that the reactor core will not reach a Departure from Nucleate Boiling Ratio (DNBR) smaller than the DNBR in the hot channel at the steady state design overpower. This corresponds to a DNBR of 1.38 at 114 per cent rated power (Table 3-1).

14.1.2.6.2 Methods of Analysis

The loss-of-coolant-flow accident is analyzed by a combination of analog and digital computer programs. Analog simulation is used to determine the reactor flow rate following loss of pumping power. Reactor power, coolant flow, and inlet temperature are input data to the digital program which determines the core thermal characteristics during the flow coastdown.

The analog model used to determine the neutron power following reactor trip includes six delayed neutron groups, control rod worth and rod insertion characteristics, and trip delay time. The analog model used to determine flow coastdown characteristics includes description of flow-pressure drop relations in the reactor coolant loop. Pump flow characteristics are determined from manufacturers' zone maps. Flow-speed, flow-torque, and flow-head relationships are solved by affinity laws.

A transient, thermal-hydraulic, B&W digital computer program is used to compute channel DNBR continually during the coastdown transient. System flow, neutron power, fission product decay heat, and core entering enthalpy are varied as a function of time. The program maintains a transient inventory of stored heat which is determined from fuel and clad temperatures beginning with the initial steady state conditions. The transient core pressure drop is determined for average channel conditions. The representative hot channel flows and corresponding DNBR are obtained by using the average core pressure drop. The hot channel DNBR as a function of time is compared with the design DNBR at maximum overpower to determine the degree of heat transfer margin.

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The loss-of-coolant-flow analysis has been carried out in the power range between 102 and 114 per cent rated power. Conditions utilized in the analysis are as follows:

a. Initial core inlet temperature for given power level is assumed to be plus 2 F in error.

b. Initial system pressure is assumed to be minus 65 psi in error.

c. Trip delay time, i.e., time for sensor detection for low flow condition until initial downward movement of control rod, is 300 milliseconds.

d. The per cent of initial reactor power as a function of time after loss of pumps is as shown in Figure 3-6.

e. The pump inertia is 70,000 $lb\text{-}ft^2$.

14.1.2.6.3 Results of Analysis

The results of this analysis show that the reactor can sustain a loss-of-coolant-flow accident without damage to the fuel. The results of the evaluation are presented in Figures 14-19 and 14-20. Figure 14-19 shows the per cent reactor flow as a function of time after loss of all pump power. Figure 14-20 shows the minimum DNBR's which occur during the coastdown for various initial power levels. The degree of core protection during coastdown is indicated by comparing the DNBR for the coastdown with the design value of 1.38 at 114 per cent rated power. This DNBR (1.38) in the representative hot channel corresponds to a 99 per cent confidence that 99.5 per cent of the core will not experience a departure from nucleate boiling under steady state conditions at the design overpower (3.2.3.1).

Under normal conditions, the maximum indicated reactor power level from which a loss-of-coolant-flow accident could occur is 102 per cent rated power (as indicated by reactor instrumentation). This power level represents an allowance of plus 2 per cent rated power for transient overshoot. This power level also represents the maximum power demand that will be permitted to the reactor control system. The 102 per cent rated power is an instrument-indicated value and is subject to the following maximum errors: (a) ±2 per cent heat balance and (b) ±4 per cent nuclear instrumentation. The true power level could be as high as 108 per cent at 102 per cent indicated power. As shown in Figure 14-20, however, the DNBR at 108 per cent is 1.44, which is significantly larger than the design DNBR.

The reactor coolant system is capable of providing natural circulation flow after the pumps have stopped. The natural circulation characteristics of the reactor coolant system have been calculated using conservative values for all resistance and form loss factors. No voids are assumed to exist in the core or reactor outlet piping. The following tabulation and Figure 9-10 show the natural circulation flow capability as a function of the decay heat generation. This material is presented in greater detail in 14.1.2.8.3.

Time After Loss of Power, sec	Decay Heat Core Power, %	Natural Circulation Core Flow Available, % Full Flow	Flow Required for Heat Removal, % Full Flow
0.36×10^2	5	4.1	2.3
2.2×10^2	3	3.3	1.2
1.2×10^4	1	1.8	0.36
1.3×10^5	1/2	1.2	0.20

The flows above provide adequate heat transfer for core cooling and decay heat removal by the reactor coolant system.

The reactor is protected against reactor coolant pump failure(s) by the protective system and the integrated control system. The integrated control system initiates a power reduction on pump failure to prevent reactor power from exceeding that permissible for the available flow. The reactor is tripped if insufficient reactor coolant flow exists for the power level. The operating limits for less than four pumps in operation have been presented in 4.3.7.

14.1.2.7 Stuck-Out, Stuck-In, or Dropped-In Control Rod

14.1.2.7.1 Identification of Cause

The control rod drives have been described in 3.2.4.3. The results of continuous control rod withdrawal have been analyzed in 14.1.2.2 and 14.1.2.3. In the event that a control rod cannot be moved because of electrical faults or mechanical seizure, localized power peaking and subcritical margin must be considered.

14.1.2.7.2 Analysis and Results

Adequate hot subcritical margin is provided by requiring a subcriticality of 1% $\Delta k/k$ subcritical with the control rod of greatest worth fully withdrawn from the core. The nuclear analysis reported in 3.2.2 demonstrates that this criterion can be satisfied.

In the event that an unmovable control rod is partially or fully inserted in the core or a single rod is dropped during operation, its location and effect on local power distribution determine whether continued power operation is permissible. The location of a stuck rod in the core will be studied further to define permissible conditions of operation. The criteria for these studies are (a) operation with a stuck rod will not increase the DNB probability above the probability specified for design conditions, and (b) a hot subcritical margin of 1% $\Delta k/k$ will be maintained with the stuck rod in its inoperative position and the operating rod of greatest reactivity worth in the fully withdrawn position.

If a control rod is dropped into the core during power operation, the same consideration of localized power peaking as for a stuck rod will apply.

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14.1.2.8 Loss of Electric Power

14.1.2.8.1 Identification of Cause

The Crystal River Plant Units 3 and 4 are designed to withstand the effects of loss of electric load or electric power. Two types of power losses are considered:

a. A "blackout" condition, caused by severe interconnected grid upset.

b. A hypothetical condition resulting in a complete loss of all Plant power.

The reactor protection criteria for these conditions are that fuel damage will not occur from an excessive power-to-flow ratio and that the reactor coolant system pressure will not exceed design pressure.

14.1.2.8.2 Results of "Blackout" Conditions Analysis

The net effect of a "blackout" condition on the nuclear units would be opening of all 230 and 500 kv breakers, thus disconnecting the Plant from the entire transmission system. When this occurs on the nuclear units, a runback signal on the integrated master controller causes an automatic power reduction to 15 per cent reactor power. Other actions that occur are as follows:

a. All vital electrical loads, including reactor coolant pumps, condenser circulating water pumps, condensate and condensate booster pumps, and other auxiliary equipment, will continue to obtain power from each unit's generator. Feedwater is supplied to the steam generators by steam-driven feed pumps.

b. As the electrical load is dropped, the turbine generator accelerates and closes the governor valves, and the reheat stop and interceptor valves. The unit's frequency will peak at less than the overspeed trip point and decay back to set frequency in 40-50 sec.

c. Following closure of the turbine governor valves and the reheat stop and interceptor valves, steam pressure increases to the turbine bypass valve set point and may increase to the steam system safety valve set point. Steam is relieved to the condenser and to the atmosphere. Steam venting to the atmosphere occurs for about 2 min. following blackout from 100 per cent rated power until the turbine bypass can handle all excess steam generated. The capacity of the modulating turbine bypass valve is 15 per cent of the valves wide open (VWO) steam flow, and that of the safety valves is 100 per cent of VWO steam flow. The first safety valve banks are set at 1,050 psig with additional banks set at pressures up to 1,104 psig (5 per cent above design pressure as allowed by code). Steam venting permits energy removal from the reactor coolant system to prevent a high pressure reactor trip. The initial power runback is to 15 per cent power which is greater than the unit's auxiliary load. This allows sufficient steam flow for regulating turbine speed control. Excess power above the unit's turbine speed control. Excess power above the unit auxiliary load is rejected by the turbine bypass valve to the condenser.

d. During the short interval while the turbine speed is high, the vital electrical loads connected to the unit generator will undergo speed increase in proportion to the generator frequency increase. All motors and electrical gear so connected are designed for the increased frequency.

e. After the turbine generator has been stabilized at auxiliary load and set frequency, the Plant operator may reduce reactor power to the auxiliary load as desired.

The blackout accident does not produce any fuel damage or excessive pressures on the reactor coolant system. There is no resultant radiological hazard to Plant operating personnel or to the public from this accident, since only secondary system steam is discharged to the atmosphere.

Unit operation with failed fuel and steam generator tube leakage is shown to be safe by the analysis presented in 11.1.2.5.2 and 14.1.2.10. For the same conditions, the steam relief accompanying a blackout accident would not change the whole body dose. The whole body dose is primarily due to the release of Xe and Kr. Release of these gases is not increased by the steam relief because even without relief, all of these gases are released to the atmosphere through the condenser vacuum pump exhaust. The rate of release of iodine during the approximately 2 min of relief would be increased by almost a factor of 10^4, because the iodine is released directly to the atmosphere rather than through the condenser and Plant vents. However, the quantity released during this short time is small, and it would be less than 0.03 MPC at the 4,400 ft exclusion distance. 1

14.1.2.8.3 Analysis Results of Complete Loss of All Plant Power

The second power loss considered is the hypothetical case where all Plant power except the Plant batteries is lost. The sequence of events and the evaluation of consequences relative to this accident are given below:

a. A loss of power results in gravity insertion of the control rods.

b. The steam generator safety valves actuate after the turbine trips and prevent excessive temperatures and pressures in the reactor coolant system.

c. The reactor coolant system flow decays without fuel damage occurring. Decay heat removal after coastdown of the reactor coolant pumps is provided by the natural circulation characteristics of the system. This capability is discussed in the loss-of-coolant-flow evaluation (14.1.2.6).

d. A turbine-driven emergency feedwater pump is provided to supply feedwater any time the main feed pumps cannot operate. The emergency feed pump takes suction from the condenser hotwell and the condensate storage. The emergency pump supplies feedwater to the

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steam generators. The emergency feed pump is driven by steam from either or both steam generators.

The controls and auxiliary systems for the emergency feed pump operate on d-c power from the Plant battery.

A recirculation line from the emergency pump discharge back to the condenser is provided to permit periodic testing

e. The condenser hotwell and the condensate storage tank provide cooling water in the unlikely event that all power is lost. The minimum condensate inventory is 200,000 gal. This inventory provides sufficient water for decay heat cooling (assuming infinite irradiation at 2,544 MWt) for a period of approximately one day.

The features described above permit decay heat cooling of the nuclear unit for an extended period of time following a complete loss of electric power.

The foregoing evaluation demonstrates the design features incorporated in the design to sustain loss of power conditions with just the Plant batteries to operate system controls. Immediate operation of the emergency feedwater pump is not of critical nature. The reactor can sustain a complete electric power loss without emergency cooling for about 25 min before the steam volume in the pressurizer is filled with reactor coolant. These 25 min are derived as follows:

a. Steam generators evaporate to dryness	10 min
b. Pressurizer safety valves open	5
c. Pressurizer fills with water (due to reactor coolant system expansion)	10
	25 min

Beyond this time reactor coolant will boil off, and an additional 90 min will have elapsed before the boiloff will start to uncover the core. The emergency feedwater pump can be actuated within this period of time. Accordingly, core protection is insured for the unlikely condition of total loss of Plant electric power.

14.1.2.9 Steam Line Failure

14.1.2.9.1 Identification of Cause

Analyses have been performed to determine the effects and consequences of loss of secondary coolant due to failures in the steam lines between the steam generators and the turbine.

The criteria for Plant protection and the release of fission products to the environment are as follows:

a. The reactor shall trip and remain subcritical without boron addition until a controlled rate of system cooldown can be effected.

b. The potential environmental consequences from radioactivity in the secondary coolant system shall not exceed those specified by 10 CFR 20.

14.1.2.9.2 Analysis and Results

The rate of reactor system cooling following a steam line break accident is a function of the area of the failure and the steam generator water inventory available for cooling. The steam generator inventory increases with power level. The inventory at rated power is 46,000 lb and decreases linearly to 20,000 lb at 15 per cent power. The steam line break accident analysis is performed at ultimate power in order to determine maximum cooling and inventory release effects. |1

The immediate effect of any steam line break accident is a reduction in steam pressure and a reduction in steam flow to the turbine. These effects initially cause the reactor control system to act to restore steam pressure and load generation.

A steam line rupture of a small area causes a relatively slow decrease in steam pressure. This places a demand on the control system for increased feedwater flow. In addition, the turbine control valves will open to maintain power generation. Increased feedwater flow causes the average reactor coolant temperature to decrease, and the resulting temperature error calls for control rod withdrawal. The limiting action in this condition is the 102 per cent limit on power demand to the rod drive control system. If the moderator temperature coefficient of reactivity is small or slightly positive, the reactor power will decrease when the control system reaches the power demand limit because of continuing temperature decrease. The reactor will then trip on low reactor coolant system pressure. A reactor trip will initiate a reduction in the feedwater flow to the steam generators.

When the moderator temperature coefficient is negative, the reactor power will tend to increase with decreasing average coolant temperature. This will cause control rod insertion to limit reactor power to 102 per cent. With power limited at 102 per cent, additional cooling causes a reduction in reactor coolant pressure, and the reactor trips on low reactor coolant pressure. Turbine trip occurs when the reactor trips. Upon turbine trip the unaffected steam line is isolated by the turbine stop valves as shown in Figure 10-1. The unit with the ruptured steam line continues to blow down to the atmosphere. |1

The maximum cooldown of the reactor coolant system would be that resulting from the blowdown from one steam generator. A typical cooling rate following reactor trip for a steam line rupture of 4 sq. in. is shown in Figure 14-21.

The tabulation below lists the approximate time required to blow down the contents of the steam generator with a ruptured steam main.

Leak Area, in.^{2}	Blowdown Time, sec
4	860
32	110
128	27

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A steam line failure of large area results in high steam flow with resulting rapid pressure decrease in the reactor coolant system and steam system. The reactor trips on low reactor coolant system pressure or high flux. Reactor trip causes turbine trip and reduction in feedwater flow to decay heat level. The turbine trip closes the turbine stop valves which isolate the steam lines 1
and prevent blowdown of the steam generator whose secondary side does not have a pipe rupture. The steam generators are designed to maintain reactor system integrity upon loss-of-secondary-side pressure. Therefore, this accident will not lead to a reactor coolant system failure.

Assuming the blowdown from one steam generator results from a secondary steam system rupture, the maximum cooling rate during this accident occurs during the first 10 sec after the break. The maximum cooling rate is approximately 3 F/sec and a low pressure or high flux trip occurs. The net cooldown of the reactor coolant system, assuming total blowdown of one steam generator and accounting for transfer of core stored heat and decay heat, is less than 50 F. This results in an average coolant temperature of 530 F which is about 10 F lower than the normal zero power average coolant temperature.

The minimum shutdown margin at 540 F with the most reactive rod stuck out is 2.9% $\Delta k/k$. The reduction in reactivity shutdown margin associated with cooling the moderator temperature 10 F below its normal shutdown temperature of 540 F would be 0.30% $\Delta k/k$. Using the maximum value for the moderator temperature coefficient (-3.0×10^{-4} $\Delta k/k/F$), the shutdown margin at 530 F would be 2.6% $\Delta k/k$, which is adequate to prevent return to criticality.

In addition, high pressure injection can be actuated during the cooldown period following a large area steam line failure. This system supplies borated water to the reactor coolant system to increase the shutdown margin further. Boron addition to the reactor coolant during the controlled cooling of the system to atmospheric pressure will prevent criticality at lower temperatures.

The effect of a steam line rupture inside the reactor building has been evaluated by conservatively assuming an instantaneous release to the reactor building of the energy associated with this accident. The mass and energy releases per steam generator in this analysis are

	Mass, lb	Energy, Btu x 10^{-6}
Steam Generator	46,000	28.0
Feedwater Flow (6 sec full flow plus coastdown to 7.5% flow @ 16 sec)	12,800	5.6
Reactor Coolant System Energy Transferred	______	17.6
Total	58,800	51.2

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Based upon the above, a single steam generator release would result in approximately 10 psig pressure rise in the reactor building. This is well below the reactor building design pressure of 55 psig.

The environmental consequences from this accident are calculated by assuming that the nuclear unit has been operating with steam generator tube leakage. The reactor coolant activity assumes prior operation with 1 per cent failed fuel rods. With these assumptions, the steam generators contain a total of 0.09 equivalent curies of iodine-131. It is further assumed that steam generator leakage continues for three hours before the nuclear unit can be cooled down and the leakage terminated. This additional leakage corresponds to 3.4 equivalent curies of iodine-131. The iodine is assumed to be released directly to the atmosphere where it mixes in the wake of the reactor building. With these assumptions an integrated dose to the thyroid at the exclusion distance of 0.53 rem is obtained. The corresponding dose to the whole body during this same time period is 0.002 rem. The total release of all activity when averaged over a year is 33 per cent of the allowable limits of 10 CFR 20.

14.1.2.10 Steam Generator Tube Failures

14.1.2.10.1 Identification of Accident

In the event of a reactor coolant leak to the secondary system, such as a complete severance of a steam generator tube, the activity contained in the coolant would be released to the secondary system. Radioactive gases and some of the radioactive iodine would be released to the atmosphere through the condenser air removal system.

14.1.2.10.2 Analysis and Results

In analyzing the consequences of this failure, the following sequence of events is assumed to occur:

a. A double-ended rupture of one steam generator tube occurs with unrestricted discharge from each end.

b. The initial leak rate, approximately 435 gpm, exceeds the normal makeup of 70 gpm to the reactor coolant system, and system pressure decreases. No operator action is assumed, and a low reactor coolant system pressure trip will occur in about 8 min.

c. Following reactor trip, the reactor coolant system pressure continues to decrease until high pressure injection is actuated at a pressure of 1,800 psig. The capacity of the high pressure injection is sufficient to compensate for the leakage and maintains both pressure and volume control of the reactor coolant system. Thereafter, the reactor is conservatively assumed to be cooled down and depressurized at the normal rate of 100 F per hour.

d. Following reactor trip, the turbine stop valves will close. Since a reactor coolant to secondary system leak has occurred, steam line pressure will increase, opening the steam bypass valves to the condenser. Each bypass valve actuates at a lower pressure than do the safety valves. The reactor coolant that leaks as a result of the tube failure is condensed in the condenser. Only the fission products that escape from the condensate are released to the atmosphere.

e. The affected steam generator can be isolated by the steam line isolation valve when the reactor coolant system pressure falls below the setpoint of the secondary system safety valves, i.e., 1,050 psig. Cooldown continues with the unaffected steam generator until the temperature is reduced to 250 F. Thereafter, cooldown to ambient conditions is continued using the decay heat removal system.

f. At the design cooling rate for the pressurizer of 100 F/hr, depressurization to 1,050 psig requires approximately 1.7 hr. During this time period 1.6×10^8 cc (5,650 ft^3) of reactor coolant leaks to the secondary system. This leakage corresponds to approximately 45,800 curies of xenon-133 if the reactor has been operating with 1 per cent failed fuel.

The radioactivity released during this accident is discharged through the turbine bypass to the condenser and then out the Plant vents. A partition factor of 10^4 is assumed for iodine in the condenser.[1,2] Noble gases are assumed to be released directly to the Plant vents. The total dose to the whole body from all the xenon and krypton released is only 0.69 rem at the 4,400 ft exclusion distance. The corresponding dose to the thyroid at the same distance is only 1.0×10^{-4} rem. This calculation conservatively assumes that the Plant vent discharge mixes in the wake of the building structures rather than remaining at its elevated release height.

14.2 STANDBY SAFEGUARDS ANALYSIS

14.2.1 SITUATIONS ANALYZED AND CAUSES

In this section accidents are analyzed in which one or more of the protective barriers are not effective and standby safeguards are required. All accidents evaluated are based on the ultimate power level of 2,544 MWt rather than the rated power level of 2,452 MWt. Table 14-3 summarizes the potential accidents studied.

Table 14-3
Situations Analyzed and Causes

Event	Cause	Effect
Fuel Handling Accidents	Mechanical damage during transfer.	Integrated dose at exclusion distance is 0.44 rem thyroid and 0.44 rem whole body.
Rod Ejection Accident	Failure of control rod drive pressure housing.	Some clad failure. Thirty-day dose at exclusion distance is 1.34 rem thyroid.
Loss-of-Coolant Accident	Rupture of reactor coolant system.	No clad melting. Thirty-day dose at exclusion distance is 9.9 rem thyroid.
Maximum Hypothetical Accident	Release of 100% rare gases, 50% iodine, and 1% solid fission products.	Two-hour dose at exclusion distance is 65 rem thyroid. Thirty-day dose at low population distance is 3.4 rem thyroid.

14.2.2 ACCIDENT ANALYSES

14.2.2.1 Fuel Handling Accidents

14.2.2.1.1 Identification of Accident

Spent fuel assemblies are handled entirely under water. Before refueling, the reactor coolant and the fuel transfer canal water above the reactor are increased in boron concentration so that, with all control rods removed, the k_{eff} of a core is no greater than 0.98. In the spent fuel storage pool, the fuel assemblies are stored under water in storage racks having an eversafe geometric array. Under these conditions, a criticality accident during refueling is not considered credible. Mechanical damage to the fuel assemblies during transfer operations is possible but improbable. This type of accident is considered the maximum potential source of activity release during refueling operations.

14.2.2.1.2 Analysis and Results

The fuel assembly is conservatively assumed to have operated at 29 MWt, twice the power level of an average fuel assembly. The reactor is assumed to have been shut down for 24 hr, which is the minimum time for reactor cooldown, reactor closure head removal, and removal of the first fuel assembly. It is
further assumed that the entire outer row of fuel rods, 56 of 208, suffers | 1
damage to the cladding. Since the fuel pellets are cold, only the gap activity is released. The fuel rod gap activity is calculated using the escape rate coefficients and calculational methods discussed in 11.1.1.3.

The gases released from the fuel assembly pass through the spent fuel storage pool water prior to reaching the auxiliary building atmosphere. As a minimum, the gases pass through 10 ft of water. Although there is experimental evidence that a portion of the noble gases will remain in the water, no retention of noble gases is assumed. Based on the data in References 3 and 4, 99 per cent of the iodine released from the fuel assembly is assumed to remain in the water. The total activity released to the building atmosphere is therefore

Iodine	28.4 curies
Noble gases	2.79×10^4 curies

The auxiliary building is ventilated and discharges through 90 per cent efficient charcoal filters to the Plant vents. The discharge from the Plant vents is assumed to mix in the wake of the building structures rather than remain at its elevated release point. This assumption produces less favorable dilution and, therefore, higher ground concentrations at the exclusion distance.

The activity is assumed to be released as a puff from the Plant vents. Atmospheric dilution is calculated using the 2-hour dispersion factor of 3×10^{-4} developed in 2.3. The total integrated dose to the whole body at the 4,400 ft exclusion distance is 0.44 rem, and the thyroid dose at the same distance is 0.44 rem. In evaluating the sensitivity of this analysis, the thyroid dose at the site boundary is directly proportional to the quantity of iodine released. For example, if only 90 per cent retention of iodine is assumed by the spent fuel storage pool water, the dose at the exclusion distance is increased by a factor of 10. The dose from this increased iodine release is still a factor of 70 below the 10 CFR 100 guidelines.

14.2.2.2 Rod Ejection Accident

14.2.2.2.1 Identification of Accident

Reactivity excursions initiated by uncontrolled rod withdrawal (14.1) were shown to be safely terminated without damage to the reactor core or reactor coolant system integrity. In order for reactivity to be added to the core at a more rapid rate, physical failure of the control rod drive housing or control rod drive nozzle must occur. Failure in the drive upper pressure housing can cause a pressure differential to act on a control rod assembly and rapidly eject the assembly from the core region. The power excursion due to the rapid increase in reactivity is limited by the Doppler effect and terminated by reactor protection system trips.

The criterion for reactor protection, should this condition occur, is that the reactor will be operated in such a manner that a control rod ejection accident will not further damage the reactor coolant system.

a. Accident Bases

The bases for the rod ejection accident are as follows:

Worth of ejected rod	0.3% $\Delta k/k$
Rod ejection time	0.150 sec
Ultimate power level	2,544 MWt
Reactor trip delay	0.3 sec

The severity of the rod ejection accident is dependent upon the worth of the ejected rod and the reactor power level. The control rod group of greatest worth is the first of the entire rod pattern to be withdrawn from the core. The worth of this rod can be as high as 30 per cent of the total pattern worth of 10.0% Δk/k, i.e., 3% Δk/k. However, 1
the 30% Δk/k value exists only when the reactor is subcritical. The details of control rod worth calculations and the methods of selecting the number of control rods in each group are presented in 3.2.2 and 7.2.2.1.2.

When the reactor is subcritical, the boron concentration is maintained at a level whereby the reactor is at least 1 per cent subcritical with the control rod of greatest worth fully withdrawn from the core. Therefore, rod ejection, when the reactor is subcritical and all other rods are in the core, does not cause a nuclear excursion As criticality is approached, the worth of the remaining control rods decreases. At criticality, rod ejection would result in a maximum reactivity addition of 0.56% Δk/k. 1

At rated power, but before equilibrium xenon is established, the total rod pattern worth remaining in the core is 2.6% Δk/k. At equilibrium xenon the pattern worth is 1.6% Δk/k. Before establishing equilibrium xenon, the greatest single control rod worth is 0.46% Δk/k. A single rod worth of up to 0.7% Δk/k has been used in the analysis of this ac- 1
cident.

In order for any one rod to have this much worth, it would necessarily be fully inserted in the core. Assuming that a pressure housing failure occurs in such a manner that it no longer offers any restriction for rod ejection, the time and therefore the rate of reactivity addition can be calculated. Further assuming that there is no viscous drag force limiting the rate of ejection, control rod travel time to the top of the active region of the core is calculated to be 0.176 sec. To account for the S-shaped reactivity worth versus position of the rod. an ejection time of 0.150 sec (75 per cent of active core height) is used in the analysis.

b. Fuel Rod Damage Criteria

Power excursions caused by reactivity disturbances of the order of magnitude occurring in rod ejection accidents could lead to three potential modes of fuel rod failure. First, for very rapid and large transients in which there is insufficient time for heat transfer from fuel to cladding, fuel melting followed by vaporization can generate destructive internal pressures without increasing cladding temperatures significantly. The second mode occurs when the internal vapor pressure is not sufficient to cause cladding rupture, but subsequent heat transfer raises the temperature of the cladding and weakens it until failure occurs. The third mode occurs when the nuclear excursion has insufficient energy to cause significant melting of the fuel. but subsequent heat transfer to clad from fuel may cause excessive cladding temperatures. In all three cases there is a possible occurrence of metal-water reactions. However, only very rapid and large transients will generate a rapid pressure buildup in the reactor coolant system.

The energy required to initiate UO_2 fuel melting is 220 to 225 cal/gm, based on an initial temperature of 68 F.[5] The heat of fusion requires an additional 60 cal/gm. Any further energy addition vaporizes the fuel and produces a buildup of vapor pressure within the fuel rod. The effect of the vapor pressure is dependent upon the temperature and ultimate strength of the cladding. Energy additions of up to 420 cal/gm have been calculated to be necessary before the bursting pressure of cladding is exceeded. The lower limit for producing significant fuel vapor pressure (14.7 psi) is 325 cal/gm.[6] The potential cladding failure is a function not only of the fuel vapor pressure, but also of fission product gas pressure, cladding and fuel irradiation exposure, and zirconium hydriding. As a lower limit, the potential for bursting of cladding and release of molten fuel to the reactor coolant is conservatively set at a fuel enthalpy of 280 cal/gm in this evaluation. 1

For power excursions with energy bursts below 280 cal/gm, zirconium-water reactions are possible. A correlation of the TREAT experiments presents a method of correlating the potential zirconium-water reaction as a function of fission energy input.[7] These data are based on initially cold (room temperature) fuel rods, but are also correlated as a function of peak adiabatic core temperature. This correlation can be used either by computing the core temperature or by adding the initial steady state fuel enthalpy to the nuclear energy burst and obtaining an equivalent final fuel enthalpy. Accordingly, a zirconium-water reaction requires a minimum fuel enthalpy of 125 cal/gm. Increasing fuel enthalpies cause a linear increase in the percentage of the reaction, which may be approximated by

$$\%Zr\text{-}H_2O \text{ Reaction} = 0.125 \text{ (Final Fuel Enthalpy - 125)}.$$

It is assumed that DNB will take place when the clad reaches a heat flux of 6.36×10^5 Btu/hr-ft^2. At this heat flux the hot fuel rod enthalpy would be approximately 140 cal/gm at EOL and 130 cal/gm at BOL. Applying the peaking factors described in 3.2.3 to the results of these analyses, the per cent of the core having an enthalpy greater than the values above can be calculated. Any fuel rod exceeding the enthalpy values above is assumed to fail from overheating and releases the gap activity of that fuel rod.

14.2.2.2.2 Method of Analysis

The hypothetical control rod ejection accident was investigated using the exact 1-dimensional WIGL2 digital computer program.[8] It was found that the point kinetics analog model results agreed with the WIGL2 results to within 10 per cent for rod worths up to 0.75% $\Delta k/k$. The point kinetics model assumes an initial flux distribution which is undisturbed by local control rod assemblies. The space-dependent model, however, has significant flux depressions in the vicinity of control rods. Although the flux throughout the core begins to increase shortly after the start of the rod ejection, the flux increase in this depressed region rises more quickly so that by the time the average power has reached a level just a few per cent above the initial power level, the flux shape has almost no perturbation in the region previously occupied by the ejected rod. The entire reactor flux then rises uniformly until the Doppler effect terminates the excursion. Thus by applying the peak-to-average flux factors of 2.92 for EOL and 3.24 for BOL to the point kinetics results, the peak and integrated flux at any point in the reactor can be accurately assessed.

14.2.2.2.3 Analysis and Results

a. Source Power

A sensitivity study at source level has been done around a single rod worth of 0.5% Δk/k. This analysis was performed with the core 0.5% Δk/k subcritical so that a total rod worth of 1% Δk/k was withdrawn in 0.150 sec. The reactor power was initially at 10^{-9} of the ultimate power level. The low pressure trip occurs at 1.7 sec after the ejection starts, and the reactor power is terminated at a peak value of 39 per cent ultimate power. This peak neutron power value is not reached until about 15 sec after the rod is ejected because Doppler feedback controls the rate of rise and magnitude of the neutron power. Therefore, a low pressure trip will terminate the accident before significant power is generated owing to the loss of coolant through the rupture. 1

An analysis was performed for the accident above without a low pressure trip to demonstrate the capability of the reactor to accept the accident.

In this case the neutron power reaches 1,000 MWt (39 per cent ultimate power), and the peak fuel temperature is 990 F. This is far below the melting temperature of UO_2, and the resultant thermal power is only 16 per cent of ultimate power. Hence, no fuel damage would result from the rod ejection accident at source power level.

b. Ultimate Power

A sensitivity study of ultimate power level has been done around an assumed single rod worth of 0.3% Δk/k. The analysis includes rod worths from 0.1 to 0.7% Δk/k, however. For the ultimate power case at beginning-of-life (BOL), the ejection of a single control rod worth 0.3% Δk/k would result in virtually no $Zr-H_2O$ reaction and approximately 1% of the core experiencing DNB (see Figures 14-22 and 14-23). The hot fuel rod would reach a peak enthalpy of about 166 cal/gm. 1

For the end-of-life case (EOL), the reactor neutron power peaks at 6,190 MWt, 200 milliseconds after the start of ejection of a 0.3% Δk/k control rod. The prompt negative Doppler effect terminates the power rise, and control rod insertion from high flux signal terminates the excursion. The total neutron energy burst during the transient is approximately 3,200 MW-sec. The final enthalpy of the nominal rod is 113 cal/gm, i.e., the enthalpy of the hot rod is 163 cal/gm. This enthalpy is considerably below the minimum range (220 to 225 cal/gm) for central fuel melting. As a result of the excursion, approximately 13.5 per cent of the core would have DNB (see Figure 14-22).

The power distribution at the beginning of core life, with the higher power peaking factors shown in 3.2.3, was used to determine the distribution of the energy of the excursion. With this zirconium cladding may react (see Figure 14-23) to contribute an additional 677 MW-sec of energy. The resultant temperature increase is spread over a relatively long period of time. Consequently, the metal-water reaction energy is liberated over a long period of time, and no damaging pressure pulses are produced in the system.

As a result of the postulated pressure housing failure, which produces a rupture size of 0.04 sq ft, reactor coolant is lost from the system. The rate of mass and energy input to the reactor building is considerably lower than that for the 3 sq ft rupture discussed in 14.2.2.3. This lower rate of energy input results in a lower reactor building pressure than that obtained for the 3 sq ft rupture.

The environmental consequences from this accident are calculated by conservatively assuming that all fuel rods that undergo a DNB result in clad failure and subsequent release of the gap activity. Actually, most of the fuel rods will recover from the DNB, and no fission product release will occur. For the case of a 0.3% $\Delta k/k$ rod ejection from ultimate power at the end of life, 13.5 per cent of the fuel rods are assumed to fail, releasing 177,000 equivalent curies of I-131 to the reactor building. Fission product activities for this accident are calculated using the methods discussed in 11.1.1.3. Using the environmental models and dose rate calculations discussed under the loss-of-coolant accident, the total integrated dose to the thyroid at the exclusion distance from this accident is only 2.68 rem in 30 days, which is more than a factor of 100 below the guideline values of 10 CFR 100.

c. <u>Sensitivity Analysis</u>

The results of a sensitivity analysis performed on the control rod ejection accident are shown in Figures 14-24 through 14-32. Figure 14-24 shows the variation in the peak neutron power as a function of the worth of the ejected control rod. For the nominal 0.3% $\Delta k/k$ case from ultimate power, the peak neutron power is less than 300 per cent, again assuming that a low pressure trip does not occur. The rod ejection from source level results in a Doppler turn-around before the flux trip is reached. Figure 14-25 shows the variation in the corresponding thermal power with control rod worth.

Figure 14-26 shows the corresponding enthalpy increase of the hot fuel rod versus control rod worth. Note the very small spread in values for the BOL and EOL ultimate power conditions. As expected, the enthalpy increases with rod worth.

Figures 14-27 through 14-30 show the peak reactor neutron and thermal powers as a function of changes in the positive moderator temperature coefficient and negative Doppler coefficient for the nominal 0.5% $\Delta k/k$ control rod ejection from source level. There was insignificant variation of the peak neutron and thermal power with changes in the two reactivity feedback coefficients.

Figure 14-31 shows the change in nominal thermal power with variations in the trip delay time for the nominal 0.3% $\Delta k/k$ rod ejection from ultimate power (the variation from zero power is negligible). The trip delay time does not affect the peak neutron power because the Doppler effect controls the power transient. Figure 14-32 shows the corresponding change in the total enthalpy increase of the hot fuel rod versus the trip delay.

The thermal power never exceeds 114 per cent ultimate power for any of the variations studied using the nominal rods (0.3% $\Delta k/k$ for ultimate power and 0.5% $\Delta k/k$ for source level). The hot fuel rod average temperature never increases by more than 310 F above the ultimate power peak value (4,090 F). It is therefore concluded that each of these parameter variations has relatively little effect on the nominal results.

14.2.2.3 Loss-of-Coolant Accident

14.2.2.3.1 Identification of Accident

Failure of the reactor coolant system would allow partial or complete release of reactor coolant into the reactor building, thereby interrupting the normal mechanism for removing heat from the reactor core. If all the coolant were not released immediately, the remaining amount would be boiled off owing to residual heat, fission product decay heat, and possible heat from chemical reactions unless an alternate means of cooling were available. In order to prevent significant chemical reactions and destructive core heatup, emergency core cooling equipment rapidly recovers the core and provides makeup for decay heat removal.

14.2.2.3.2 Accident Bases

All components of the reactor coolant system have been designed and fabricated to insure high integrity and thereby minimize the possibility of their rupture. The reactor coolant system, the safety factors used in its design, and the special provisions taken in its fabrication to insure quality are described in Section 4.

In addition to the high-integrity system to minimize the possibility of a loss of coolant, emergency core cooling is provided to insure that the core does not melt even if the reactor coolant system should fail and release the coolant. This emergency core cooling is provided by the core flooding system, the makeup and purification system (high pressure injection), and the decay heat removal system (low pressure injection). These systems are described in detail in Section 6, and their characteristics are summarized in the following paragraphs.

The performance criterion for the emergency core cooling equipment is to limit the temperature transient below the clad melting point so that fuel geometry is maintained to provide core cooling capability. This equipment has been conservatively sized to limit the clad temperature transient to 2,300 F or less as temperatures in excess of this value promote a faster zirconium-water reaction rate, and the termination of the transient near the melting point would be difficult to demonstrate. | 1 | 1

The fuel rods may experience cladding failure during the heatup in the loss-of-coolant accident. This could be due to fission gas internal pressure and weakening of the clad due to the increase in clad temperature. The mechanical strength of the Zircaloy cladding is reduced as the temperature exceeds 1,000 F such that the highly irradiated fuel rods, with high fission gas internal pressure, may fail locally and relieve the gas pressure when the temperature exceeds 1,200 F. Some local ballooning of rods is likely to occur. However, cooling would still be effective since the fuel rods are submerged, and cross-channel flow around the ballooned area will cool the rod. At worst a local hot spot may occur.

It is calculated that a small number of fuel rods operating at peak power will experience a cladding temperature transient to 1,950 F in about 18 sec. The injection of emergency coolant, at a time when the cladding is at a temperature of about 1,950 F, may also cause distortion or bowing between supports. As a result some of the fuel rods may crack and allow relief of internal pressure.

However, the cladding is expected to remain sufficiently intact to retain the solid fuel material and to prevent gross fuel shifting. The transient would be limited to regions of the core which operate at peak power. The major portion of the core will not experience as severe a transient.

Heating of the fuel can and the fuel rod spacer grid requires heat flow from the clad to the structure by conduction and radiation; therefore, the structure temperatures will lag the cladding temperature transient. As the fuel rod temperature rises, the fuel rods are expected to experience some bowing between supports due to the temperature differential existing between the fuel rod and the can. The cans and spacer grids are made from stainless steel and have substantial mechanical strength, even at the maximum expected temperatures. The supporting stainless steel structure will therefore retain sufficient strength to assure spacing between fuel rods to allow emergency coolant to reach them, and will keep the fuel rods in the same location in the core to prevent gross fuel shifting.

The core flooding system has two independent core flooding tanks, each of which is connected to a different reactor vessel injection nozzle by a line containing two check valves and a normally open, remotely operated isolation valve. Since these tanks and associated piping are missile-protected and are connected directly to the reactor vessel, a rupture of reactor coolant system piping will not affect their performance. These tanks provide for automatic flooding when the reactor coolant system pressure decreases below 600 psi. The flooding water is injected into the reactor vessel and directed to the bottom of the reactor vessel by the thermal shield. The core is flooded from the bottom upward. The combined contents of the two tanks (1,880 ft^3 of borated water) rapidly reflood the core immediately after the blowdown to provide cooling until coolant flow can be established by low pressure injection.

High pressure injection, actuated by low reactor coolant system pressure, sup-
plies coolant at pressures up to the design pressure of the reactor coolant sys-
tem and at a rate up to 1,000 gpm. Low pressure injection actuated by low reac- 2
tor coolant system pressure supplies coolant at pressures below 100 psig and at
a rate up to 6,000 gpm. Both of these systems can operate at full capacity from 2
the on-site emergency electrical power supply and can be in operation within 25
sec after the accident. In the reactor vessel, decay heat is transferred to the
injection water.

Injection water is supplied from the borated water storage tank. When this tank empties, water is circulated from the reactor building sump through heat exchangers and returned to the reactor vessel.

Engineered safeguards are also provided to cool the reactor building environment following a loss-of-coolant accident and thereby limit and reduce pressure in the building. Reactor building sprays, actuated on a high building pressure signal of 10 psig, deliver 3,000 gpm to the reactor building atmosphere. This spray water reaches thermal equilibrium within the building atmosphere during its passage from the nozzles to the sump. Spray water is supplied from the borated water storage tank until it is emptied. Thereafter, water collected in the sump is recirculated to the sprays. Cooling is also provided by the reactor building emergency cooling system in which recirculating fans direct the steam-and-air mixture through emergency coolers, where steam is condensed. Heat absorbed in the emergency coolers is rejected to the nuclear services cooling water system. The heat removal capacity of either of these two reactor building cooling systems is adequate to prevent overpressurization of the building during a loss-of-coolant accident.

This analysis demonstrates that in the unlikely event of a failure of the reactor coolant system, both of the other two boundaries that prevent fission product release to the atmosphere, i.e., the reactor core and the reactor building, are protected from failure. Accordingly, the public would be protected against potential radiation hazards.

In order to evaluate this accident, a range of rupture sizes from small leaks up to the complete severance of a 36 in. ID reactor coolant system line has been evaluated. A core cooling analysis is presented for the complete severance of the 36 in. ID reactor coolant piping.

Since the large rupture removes the least amount of stored energy from the core, this represents the minimum temperature margin to core damage and, therefore, places the most stringent requirements on the core flooding system.

The reactor building pressures have been evaluated for a complete spectrum of rupture sizes without the action of core flooding tanks and, therefore, are conservative. The peak pressure occurs for a 3 ft^2 rupture rather than for a 36 in. ID (14.1 ft^2) rupture. Rupture sizes smaller than the 36 in. ID leak result in longer blowdown times, and the amount of energy transferred to the reactor building atmosphere is increased. As a result the intermediate leak size results in a reactor building pressure greater than that produced by the 36 in. ID rupture.

14.2.2.3.3 Accident Simulation

a. Hydraulic Model

Blowdown of the reactor coolant system following an assumed rupture has been simulated by using a modified version of the FLASH[9] code. This code calculates transient flows, coolant mass and energy inventories, pressures, and temperatures during a loss-of-coolant accident. The code calculates inflow from the emergency cooling and calculates heat transferred from the core to the coolant.

Modifications were made to FLASH to make the model more applicable to this system. The changes are as follows:

(1) The calculation of reactor coolant pump cavitation was based on the vapor pressure of the cold leg instead of the hot leg water.

(2) Core flooding tanks have been added. Water flow from the core flooding tanks is calculated on the basis of the pressure difference between the core flooding tanks and the point of discharge into the reactor coolant system. The line resistance and the inertial effects of the water in the pipe are included. The pressures in the tanks are calculated by assuming an adiabatic expansion of the gas above the water level in the tank. Pressure, flow rate, and mass inventories are calculated and printed out in the computer output.

(3) Additions to the water physical property tables (mainly in the subcooled region) have also been made to improve the accuracy of the calculations.

(4) A change in the steam bubble rise velocity has been made from the constant value in FLASH to a variable velocity as a function of pressure. The bubble velocity term determines the amount of water remaining in the system after depressurization is complete. For large ruptures, this change in velocity shows no appreciable change in water remaining from that predicted by the constant value in the FLASH code. For smaller ruptures, an appreciable difference exists. The variable bubble velocity is based on data in Reference 10 and adjusted to correspond to data from the LOFT semiscale blowdown tests.

Test No. 546 from the LOFT semiscale blowdown tests is a typical case for the blowdown through a small rupture area. A comparison of the predicted and experimentally observed pressures is shown in Figure 14-33. Figure 14-34 shows the per cent mass remaining in the tank versus time as predicted by the code. At the end of blowdown, the predicted mass remaining is 13 per cent. The measured mass remaining is approximately 22 per cent.

The FLASH code describes the reactor coolant system by the use of two volumes plus the pressurizer. The system was grouped into two volumes on the basis of the temperature distribution in the system as follows:

Volume 1 includes half of the core water volume, the reactor outlet plenum, the reactor outlet piping, and approximately 55 per cent of the steam generators.

Volume 2 includes half of the core water volume, the reactor inlet plenum and downcomer section, the reactor inlet piping, pumps, and 45 per cent of the steam generators.

Volume 3 represents the pressurizer.

The resistances to flow were calculated by breaking the reactor coolant system into 24 regions and calculating the volume-weighted resistance to flow for a given rupture location based on normal flow resistances. For the double-ended ruptures, all of the leak was assumed to occur in the volume in which that pipe appeared.

The reactor core power was input as a function of time as determined by the CHIC-KIN code in conjunction with the FLASH output. Steam generator heat removal was assumed to cease when the rupture occurred.

While the modified FLASH code now has the capability of simulating injection flow from the core flooding tanks, the calculations shown in this report were made prior to the time that the core flooding simulation was added to FLASH. Core flooding was calculated by taking the reactor vessel pressure as predicted by FLASH without core flooding and using this pressure as input to a separate program to get the flow from the core flooding tanks. Reactor vessel filling

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was calculated by adding the mass remaining in the vessel as predicted by FLASH to the mass injected from the core flooding tanks. This method of calculation is conservative in that condensation of steam by the cold injection water is not taken into account. A more recent analysis using the FLASH code confirms that conservatism used in this analysis.

Pressure, temperature, mass and energy inventories, and hydraulic characteristics as determined by FLASH are input into the core thermal code (SLUMP) and the reactor building pressure buildup code (CONTEMPT).

b. Core Thermal Model

The core heat generation and heat transfer to the fluid are dependent upon the blowdown process. The FLASH program includes a core thermal model and the feedbacks of heat transfer and flow on each other. While the FLASH thermal model is acceptable for determining the effect of core heat transfer on the blowdown process, a more extensive simulation is necessary for evaluation of the core temperature transient.

Additional analytical models and digital computer program (SLUMP) were developed to simulate the core thermal transient for the period beginning with the initiation of the leak and ending after the core temperature excursion had terminated.

The model includes the effects of heat generation from neutrons before reactor trip, neutron decay heat, and fission and activation product decay heat; the exothermic zirconium-water reaction based on the parabolic rate law; heat transfer within the fuel rods, limited heat convection from the fuel clad surface to any fluid within the core region, heat transfer from reactor vessel walls and internals to the coolant, and heat transfer from fuel rods to the steam necessary to sustain a metal-water reaction; and emergency injection flow and boiloff.

The basic model structure provides 50 equal-volume core regions with input provisions to allow any choice of power distribution. The model may be used to simulate the entire core or any subdivision of the core. Therefore, the core geometry may be detailed to the degree consistent with the results desired.

The following parabolic law for the zirconium-water reaction equation (11) with the following constants is simulated for each of the regions:

$$- \frac{dr}{dt} = \frac{K}{(r_o - r)} \quad \exp - \frac{\Delta E}{RT}$$

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where

r = radius of unreacted metal in fuel rod

r_o = original radius of fuel rod

t = time

K = rate law constant (0.3937 cm^2/sec)

ΔE = activation energy (45,500 cal/mole)

R = gas constant (1.987 cal/mole K)

T = temperature, K

The zirconium-water reaction heat is assumed to be generated completely within the clad node. The heat necessary to increase the steam temperature from the bulk temperature to the reaction temperature is transferred from the clad at the point of reaction. The above equation implies no steam limiting. However, the program does have provision for steam rate-limiting to any degree desired, but no steam-limiting of the reactions has been assumed. All heat from neutron, beta, and gamma sources is assumed to be generated within the fuel according to the preaccident power distribution and infinite irradiation.

Within each of the regions there is a single fuel node and a single clad node with simulation of thermal resistance according to the normal fuel rod geometry. Provision is made to simulate four different modes of heat transfer from the clad node to the fluid sink node by specifying the time-dependent surface coefficient.

The surface heat transfer coefficient input data are determined from calculations which are based on flow and water inventory as furnished from the blowdown and the core flooding tank performance analysis.

In the event that insufficient cooling is provided, the program will allow clad heating to progress to the melting point. At this point the latent heat of zirconium must be added before the clad melts. Provisions are also incorporated to allow the clad to be heated to temperatures above the melting point before slump occurs.

As each region slumps it may be assumed to surrender heat to a water pool or to some available metal heat sink. If water is available an additional 10 per cent reaction is assumed to occur.

The program output includes the following (as a function of time unless otherwise specified):

Average fuel temperature of each region.

Average clad temperature of each region.

Per cent metal-water reaction in each region.

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Time for the clad of each region to reach the metal-water threshold, the beginning and end of melting, and the slump temperature.

Heat transferred to the reactor building from the core.

Heat generation by hydrogen and oxygen recombination.

Total zirconium-water reaction.

Total heat stored in metal sinks.

c. Reactor Building Pressure Model

The reactor building pressure-temperature analysis is performed using the digital computer code "CONTEMPT" developed by Phillips Petroleum Company in conjunction with the LOFT project. This program and its capabilities are described in Reference 12. With minor modifications this program was adapted for use on the B&W Philco-2000 computer.

In this model, the reactor building is divided into two regions: the atmosphere (water vapor and air mixture) and the sump region (liquid water). Each region is considered to be well mixed and in thermal equilibrium, but the temperature of each region may be different. The reactor building and its internal structures are subdivided into five segments, as shown in Table 14-4, and treated as slabs with 1-dimensional heat transfer. Each segment, divided into several heat conducting subregions, may act as a heat source or sink. The program includes the capability of cooling the reactor building atmosphere by air coolers (reactor building emergency cooling units) and spray coolers (reactor building spray system), and cooling the liquid region by sump coolers (decay heat removal coolers).

During blowdown, mass and energy are added directly to the atmosphere where the liquid water present is assumed to fall to the liquid region. After blowdown is over and emergency injection has been initiated, mass and energy are also added directly to the vapor region as steam. When the water level in the reactor vessel reaches the nozzle height, all mass and energy are added directly to the liquid region since no boiling of the injection water occurs after the core has been covered. When the supply of injection water is depleted, recirculation and cooling of sump water is simulated.

The reactor building calculations are begun by computing steady-state results using initial atmospheric conditions as the input. Following the rupture, the mass and energy addition is determined from the energy input rates for each time step. Heat losses or gains due to the heat-conducting slabs are calculated. Then the pressure and temperature of the liquid and vapor regions are calculated from the mass, volume, and energy balance equations.

Table 14-4
Reactor Building Structural Heat
Capacitance Segments

Segment	Description
1	Reactor Building Walls and Dome
2	Refueling Cavity (Type 304 SS Liner - One Side)
3	Reactor Building Floor
4	Internal Concrete
5	Internal Steel

The model has been developed so that the effectiveness of the natural heat sinks and the engineered safeguards can be clearly demonstrated. The model can readily produce the reactor building pressure history for any assumed combination of operable safeguards. Therefore, the effectiveness of any given arrangement can be analyzed.

14.2.2.3.4 Accident Analysis

a. Core Flooding Tank Design Base Accident

The 36 in. ID, double-ended pipe rupture produces the fastest blowdown and lowest heat removal from the fuel. This case therefore represents the most stringent emergency core cooling requirements. Results from the modified version of FLASH indicate that the core flooding tank simulation provides for the retention of all injection plus a portion of the original reactor coolant that would otherwise have been released. Thus, the cool injection water provides a cooling and condensing effect which reduces overall leakage. For the present analysis, no credit has been taken for the extra accumulation of water due to the condensing effect.

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The SLUMP digital computer program, as described in 14.2.2.3.3.b above, is used to evaluate core flooding tank performance in terms of core cooling capability. In the analysis, the hottest 5 per cent of the core was simulated in segments of 1/4 of one per cent each. The hottest segment was assigned a peaking factor of 3.1 times the average of the total core power density.

A detailed analysis of the void shutdown and core response was made with the digital computer program CHIC-KIN. This program accounts for changes in flow, pressure, enthalpy, and void fraction. It also computes axially weighted Doppler and moderator coefficients of reactivity for the kinetics calculation. The Doppler coefficient is input as a nonlinear function of fuel temperature, and the moderator void coefficient is input as a function of void fraction. The parameters describing the coolant were obtained from the digital computer program FLASH, which in turn used the neutron power output from CHIC-KIN. The core is assumed to be initially at the ultimate power level of 2,544 MWt. 1

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Figures 14-34a and 14-34b show the results of the hot leg, 14.1-ft^2 rupture simulation without trip action. Figure 14-34a is the neutron power trace, and Figure 14-34b shows the various components of the reactivity feedback. 1

The transient core flow from the FLASH analysis of the 36 in. ID, double-ended rupture was used to determine the core cooling mechanism used in SLUMP. The very high flow rates during the initial blowdown period provide nucleate boiling conditions. However, the time for Departure from Nucleate Boiling (DNB), especially for the hot regions, is extremely difficult to determine. Therefore, a conservative approach was adopted by assuming DNB at 0.25 sec. Nucleate boiling surface coefficients at high flow rates may exceed 50,000 Btu/hr-ft^2-F. A nucleate boiling surface coefficient of 25,000 Btu/hr-ft^2-F was used in the analysis. However, the series heat transfer from the clad node to the fluid sink is limited to 6,500 Btu/hr-ft^2-F by the relatively low conductance of the clad.

After DNB the surface heat transfer was calculated using the flow provided by FLASH results and Quinn's modified version of the Sieder-Tate[13] correlation:

$$h_{TPF} = 0.023 \frac{k}{D_h} (N_{Re})^{0.8} (N_{Pr})^{1/3} \left[1 + \frac{1 - X}{X}\left(\frac{\rho_B}{\rho_F}\right)\right]^{0.8} \left(\frac{\mu_B}{\mu_W}\right)^{0.14}$$

where

h_{TPF} = two-phase film heat transfer coefficient, Btu/hr-ft^2-F

k = fluid conductivity, Btu/hr-ft^2-F

D_h = hydraulic diameter, ft

N_{Re} = Reynolds number

N_{Pr} = Prandtl number

x = quality

ρ = density

μ = viscosity

subscript B = "Bulk"

subscript F = "Film"

subscript W = "Wall"

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With this correlation, bulk steam properties are used in the basic form, and the last two bracketed terms are modifiers which correct for quality and different conditions at the wall.

Figure 14-35 shows the core flow vs time for the 14.1 ft^2 leak as calculated by FLASH.

Figure 14-36 shows the clad surface heat transfer coefficient versus time based on the flow of Figure 14-35 and the modified Sieder-Tate equation. The straight line in Figure 14-36 indicates the surface heat transfer values which were used in SLUMP, and which are conservative as compared to the results obtained from the Sieder-Tate equation.

In applying the Sieder-Tate equation constant values of bulk steam quality and temperature corresponding to the most conservative assumptions were used.

A sensitivity analysis was made for maximum coefficients in SLUMP ranging from 400 to 2,000 Btu/hr-ft^2-F initially and decreasing to zero at the end of blowdown. Results are discussed below.

After blowdown no core cooling is assumed until after core recovering starts. When the water level reaches the core bottom and starts to rise up on the core, the submerged portion will be cooled by pool boiling, and any steam thus produced will provide some cooling for that portion of the core above the water line. However, in determining peak clad temperatures no cooling is assumed for that portion of the core which is above the water line.

.. the point of initial contact of cool water against hot cladding the heat flux and temperature differences will be such that film boiling is the probable mode of heat transfer. This mode provides the lowest surface coefficients which would be in the range of 100 to 300 Btu/hr-ft^2-F. However, in evaluating the core flooding tank design a conservative approach was used by assuming a value of 20 Btu/hr-ft^2-F. This value is adequate for terminating the temperature excursion in the clad.

The core flooding tank analysis incorporated the study of performance sensitivity to three significant core flooding tank parameters: (a) gas pressure (400 to 1,000 psig), (b) ratio of nitrogen gas volume to total volume (1/3 and 1/2), (c) and size of piping between the core flooding tanks and the reactor vessel (12 in. and 14 in. ID). Figure 14-37 shows the reactor vessel water level versus time for core flooding tanks operating at 600 psig with different combinations of volume ratio and line size. This figure includes an allowance for boiloff and also shows the effect of the flow provided by high pressure and low pressure injection beginning at 25 sec when emergency power is available. Similar curves for 400 psig and 1,000 psig core flooding tanks are shown in Figure 14-38. Figure 14-39 shows the maximum clad temperature reached by the hot spot and by the 1, 2, 3, 4 and 5 percentiles of the core as a function of quench time.

The quench time for a given percentile is taken as that time when the water level reaches the highest point in the core at which the peaking factor corresponding to that percentile exists. The fact that the same peaking factor may exist at some lower point in the core provides an inherent conservatism in the data as plotted. The axial peaking factor profile for the beginning of core life was used as it imposes the most stringent requirements on the core flooding tank design.

Peak clad temperatures for the core flooding systems described above are also shown on Figure 14-39. These curves demonstrate that all of the systems presented are capable of keeping the peak temperature at the hot spot more than 1,000 F below the melting temperature of the clad. The amount of zirconium-water reaction which occurs for each of these core flooding systems is shown in Table 14-5. While this preliminary analysis indicates some difference in the performance of the systems, it is not considered to be a significant difference since the analysis was performed without considering the effects of condensation by the core flooding coolant or of possible bypass to the leak of part of the coolant. 2

The preliminary core flooding tank design selected is for a 600 psi charge pressure, 940 ft^3 of water, 470 ft^3 of nitrogen, and a 14 in. supply line. The performance of this system in limiting core temperatures is approximately in the center of the range for the systems described. The parameters selected for the final system design will be based on the results of core melting analyses to be conducted as part of the final design of the reactor. For this 600 psi charge pressure, Figure 14-39 indicates that the hot spot clad temperature would reach 1.950 F at 17.5 sec and that less than 5 per cent of the core would exceed 1,690 F. For this same case calculations indicate less than 0.005 per cent total zirconium-water reaction for the whole core.

Table 14-5
Core Flooding Tank Performance Data

Pressure	Line Size, in.	Nitrogen Volume, % of Total	Total Metal Water Reaction, %
400	14	33	.022
400	14	50	.009
600	14	33	.005
600	14	50	.002
600	12	33	.022
600	12	50	.010
1,000	12	33	.003
1,000	12	50	0

Additional analysis was performed to evaluate the sensitivity of the maximum clad temperature to three important thermal parameters. All cases discussed below have in common the following parameters:

Leak size:	14.1 ft^2
Time of DNB:	0.25 sec
Time at ultimate power:	2 sec
Time that blowdown cooling ends:	9.5 sec
Core region:	Hot spot
Time to initiate quenching:	18 sec
Dependent variable examined:	Clad temperature for hottest 5 per cent of core.

Figure 14-40 shows the clad maximum temperature sensitivity to the initial surface heat transfer coefficient after the 0.25 sec nucleate boiling period. The coefficient is linearly decreased to zero at 9.5 sec. Zero cooling is maintained until quenching is initiated with a clad surface coefficient of 20 Btu/hr-ft^2-F. Previous discussion indicated justification for assuming 1,000 Btu/hr-ft^2-F for the clad surface at 0.25 sec. Figure 14-40 shows that a value of 1,000 is not on the most sensitive part of the curve and a 20 per cent decrease in h would only result in increasing the peak clad temperature 120 F.

The assumption that DNB occurs at 0.25 sec is quite conservative. 1
The duration of the nucleate boiling period has been evaluated to show the sensitivity of the maximum fuel temperature to this parameter. Figure 14-44 shows the effect of variation of time to reach a DNB. It should be noted that if DNB occurred at the time of rupture, the peak temperature would only increase about 30 F above 1,950 F.

Figure 14-41 shows hot spot clad temperature transients for a range of injection cooling coefficients. All cases have a clad surface coefficient of 1,000 Btu/hr-ft^2-F at 0.25 sec, decreasing to zero at 9.5 sec. Heat removal is then zero until the effect of injection cooling is simulated. Figure 14-41 shows that without any cooling the temperature reaches the melting point in approximately 50 sec.

The analysis of core cooling has been based upon 2.1 full-power sec- 1
onds resulting from a void shutdown using the maximum positive moderator temperature coefficient of $+1.0 \times 10^{-4}$ $(\Delta k/k)/F$. The effect of variation of the integrated power on hot spot clad temperature is shown in Figure 14-46. The resultant integrated power before a void shutdown occurs could increase to 3.4 full-power seconds before the hot spot clad temperature would reach 2,300 F, the temperature at which 1.0 per cent Zr-water reaction occurs.

An h value of 15 stops the fast temperature excursion and allows only a low rate of increase thereafter. Since the continuously increasing depth of coverage provided by the flooding tanks and the pumped flow injection systems provide additional cooling capability with time, an initial cooling value as low as 15 is probably adequate.

An h value of 20 provides immediate quenching action and a slow cooling rate thereafter.

An h value of 100 provides very fast cooling. Even though the 100 is a realistic value for film boiling in a pool - the probable mode for the submerged portion of the core - a more conservative value of 20 has been used as the reference for evaluating core flooding tank performance.

Figure 14-42 shows hot spot clad temperature transients for a range of pool fluid sink temperatures. Parameters for heat transfer prior to 18 sec are the same as discussed in the preceding paragraph. At 18 sec a surface coefficient of 20 Btu/hr-ft^2-F was applied with sink temperatures as indicated. All results reported herein previously have had a sink temperature of 280 F during the quenching period. Prior to quenching the sink temperature in all cases is based on the transient fluid pressure which results from the FLASH analysis. Figure 14-42 shows that any sink temperature below approximately 500 F is adequate for holding or reducing the clad temperature which existed at 18 sec. The core flooding tanks will provide a high flow of cool water. Although some heating will occur from contact with hot metal before the injection water reaches the core, the temperature rise could not be over 50 F assuming that the water came in contact with all reactor coolant system metal below the nozzle level before it contacted the core. Using a reference value of 280 F provides an added conservatism to the analysis.

In conclusion, the analysis has shown that the preliminary design of the core flooding system will provide for covering approximately 80 per cent of the core at 25 sec after the double-ended rupture of the 36 in. ID pipe first occurs. Beyond this time high pressure and low pressure injection will provide a continuous increase in the water level.

The clad hot spot temperature excursion is terminated at 1,950 F and less than 5 per cent of the total cladding exceeds 1,690 F. Only a minute amount (0.005 per cent) of zirconium-water reaction occurs, and the maximum temperature is at least 1,400 F below the clad melting point.

The temperature transient in the core can produce significantly higher than normal temperatures in components other than fuel rods. Therefore a possibility of eutectic formation between dissimilar core materials exists. Considering the general area of eutectic formation in the entire core and reactor vessel internals, the following dissimilar metals are present, with major elements being in the approximate proportions shown.

Type 304 Stainless Steel

19 per cent Chromium
10 per cent Nickel
Balance Iron

Control Rod

80 per cent Silver
15 per cent Indium
5 per cent Cadmium

Zircaloy-4

98 per cent Zirconium
1-3/4 per cent Tin

UO_2

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All these elements have relatively high melting points, i.e., greater than 2,700 F, except those for silver, cadmium, and indium which, in the case of indium, is as low as approximately 300 F.

The binary phase diagram indicates that zirconium in the proportion of 75 to 80 per cent has a eutectic point with either iron, nickel, or chromium at the temperatures of approximately 1,710, 1,760, and 2,370 F, respectively. If these dissimilar metals are in contact and if these eutectic points are reached, the materials could theoretically melt even though the temperature is below the melting point of either material taken singularly.

One point of such dissimilar metal contact is between Zircaloy-clad fuel rods and stainless steel spacers. The analysis of the performance of the core flooding tanks during a loss-of-coolant accident indicated that only 4 per cent of the cladding would ever exceed the zirconium-iron eutectic point. Since the spacers are located at 21 in. intervals along the assembly and each [illegible] has a very small contact area, only a fraction of the 4 per cent would be in contact with stainless steel. The approximate time period that the 4 per cent of the cladding is above the eutectic point is 30 sec. Because of the relatively small area of contact, the condition could not progress very far and fuel geometry would be maintained. Unless the proper ratio of metals is available, the melting point is higher than the eutectic point.

Another area of dissimilar metal contact is that of a zirconium guide tube with the stainless steel cladding of the control rod. Following blowdown, heat can be generated in the control rods by absorption of gamma rays. Beta ray decay heat will be deposited in the fuel rods where generated. Since gamma decay heat is only about one-half the total decay heat, and the control rod is shielded from the fuel by a guide tube, heat generation rates in control rods will be less than one-half the rates in the fuel. As a result, the peak heat generation rate in control rods adjacent to hot spot fuel would not exceed an estimated one-half times the rate in these fuel rods which have a 3.1 power ratio. The contribution from radiant heat transfer from higher powered fuel rods would be relatively small. The analysis of core melting shows that, with core flooding tanks, fuel rods with a 1.5 power ratio will not exceed 1,500 F. This is well below the eutectic melting point.

The reactor core will remain subcritical after flooding without control rods in the core because the injection water contains sufficient boron (2,270 ppm) to hold the reactor subcritical at reduced temperatures. The most stringent boron requirement for shutdown without any control rods is at the beginning of core life when the reactor is in a cold, clean condition and 1,820 ppm boron are required to maintain | 1
k_{eff} of 0.99. (See Table 3-6, Soluble Boron Levels and Worth.) The concentration existing in the reactor building sump after a loss-of-coolant accident from operating power at the beginning of core life is 2,174 ppm boron. This concentration represents a boron margin of | 1
354 ppm above the subcriticality design value margin of 1 per cent.

b. <u>Core Cooling Analysis for Spectrum of Leak Sizes</u> 1

An analysis of the loss-of-coolant accident has been made for a spectrum of leak sizes and locations. This information has been analyzed and is reported according to the following grouping: (1) hot leg ruptures, (2) cold leg ruptures (3) injection line failures, and (4) injection system capability.

(1) <u>Hot Leg Ruptures</u>

In 14.2.2.3.4a an analysis of the 36-in. ID, double-ended pipe rupture was presented. This accident produced the fastest blowdown and lowest heat removal from the fuel, therefore producing the highest cladding temperatures of any loss-of-coolant accident. This was therefore the basis for design of the core flooding equipment. A decrease in the rupture size assumed results in decreased maximum clad temperature during a loss-of-coolant accident.

Core cooling evaluations have been performed for a spectrum of four additional rupture sizes using the same basic calculational technique and assumptions as for the large rupture case. These rupture sizes are 8.5, 3.0, 1.0, and 0.4 ft^2. The reactor coolant system pressure-time history for these rupture sizes is shown in Figure 14-44.

The reactor vessel water volume as a function of time after the rupture for the various rupture sizes is shown in Figure 14-44-a. These water volume curves were generated utilizing the flow available from core flooding tanks, one high pressure injection 5
pump, and one low pressure injection pump. The pumping systems were assumed to have a combined capacity of at least 3,500 gpm with the high pressure pump running on emergency power within 25 sec after the rupture, and the low pressure pump delivering 3,000 gpm when the pressure has decayed to 100 psi, or at 25 sec, whichever occurs later.

Figure 14-44-b shows the hot spot clad temperature as a function of time for the various rupture sizes. As can be seen from this figure, the small-sized ruptures yield maximum clad temperatures which are considerably lower than those resulting from the larger sizes. The results of this study are shown in the following Table 14-5-1.

Table 14-5-1
Tabulation of Loss-of-Coolant Accident Characteristics
for Spectrum of Hot Leg Rupture Sizes

Rupture Size, ft^2	Full-Power Seconds	Min. Water Level Below Bottom of Core, ft	Hot Spot Max. Temp., F
14.1	2.1	-6.8	1,950
8.5	3.4	-5.2	1,916
3.0	1.5(*)	-2.2	1,235
1.0	1.5(*)	+4.7	1,075
0.4	1.5	+12.0	1,015

(*) Blowdown forces on control rods are equal to, or less than, normal pressure drop, and control rods will insert with normal velocities. These values are for trip shutdown rather than for a void shutdown, but include void reactivity effects.

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(2) Cold Leg Ruptures

A similar analysis of a spectrum of rupture sizes has been made for the cold leg piping. The rupture sizes tabulated are the double-ended, 28-in. ID inlet pipe, which yields 8.5 ft^2 of rupture area, and the 3.0, 1.0 and 0.4-ft^2 sizes.

The reactor coolant system average pressure for this spectrum of rupture sizes as a function of time is shown in Figure 14-44-c. The water level as a function of time is shown on Figure 14-44-d. The water level calculation has been based upon uninhibited flooding as the check valves are provided in the core support barrel to equalize pressures and permit the trapped steam above the core to escape out the rupture.

The hot spot temperature as a function of time for the spectrum of cold leg leak sizes is shown in Figure 14-44-e. The results of this analysis are shown in the following Table 14-5-2.

Table 14-5-2
Tabulation of Loss-of-Coolant Accident Characteristics
for Spectrum of Cold Leg Rupture Sizes

Rupture Size ft^2	Full-Power(*) Seconds	Min. Water Level Below Bottom of Core, ft	Hot Spot Max. Temp., F
8.5	0.4(*)	-6.7	1,785
3.0	1.0(*)	-4.8	1,575
1.0	1.3(*)	+3.6	1,250
0.4	1.3	+7.0	1,090

(*) Blowdown forces on control rods are equal to, or less than, normal pressure drop, and control rods will insert with normal velocity. These values are for trip shutdown rather than void shutdown, but include reactivity effects.

(3) Evaluation of Emergency Coolant Injection Line Failure 1

The evaluation of a low pressure injection line failure has been made, and the results of the analysis show that the reactor is protected. The rupture of a pipe which connects a core flooding tank and the low pressure injection flow to the reactor vessel was assumed to fail adjacent to reactor vessel and before the first check valve. (See Figure 6-1.) This pipe has an internal diameter of 11.5 in., and the resultant rupture area is 0.72 ft^2.

Interpolation of available blowdown calculations has been used to evaluate this rupture size, and the data show that a rupture of this size would result in the core being uncovered several feet below the top of the core. However, the hot spot will never be uncovered, and peak cladding temperatures will be slightly less than that shown in Figure 14-44-e for the 1.0 ft^2 cold leg rupture.

Since this small rupture size leaves a considerable water inventory in the reactor vessel, the remaining core flooding tank inventory is more than adequate to completely reflood the core.

The other low pressure system can supply 3,000 gpm of water to 5
the reactor vessel and provide coolant to keep the core cooled. The combined capacity of the two high pressure pumps is 1,000 gpm which is in excess of the boiloff rate (680 gpm) due to decay heat immediately after blowdown. With a single 500 gpm high pressure injection pump the excess water above the core is adequate to prevent the core from being uncovered below the three quarter elevation and beyond 300 sec the water level will begin to increase.

The high pressure injection system has two independent chains of flow to supply borated coolant to the system. If a rupture of high pressure injection piping were to occur in one of the four lines between the attachment to the primary pipe and the check valve, the other chain of this system would have adequate capacity to protect the core against this small leak. In the event of a component failure in the second high pressure injection loop, the ruptured flow path can be monitored by the operator and spillage flow can be stopped by isolation of the affected piping. The entire capacity of one pump can then be utilized to handle the small rupture and protect the core.

(4) Evaluation of Emergency Core Injection System 1
Performance for Various Rupture Sizes

The loss-of-coolant analysis is based upon the operation of one 5
high pressure injection pump (500 gpm), one low pressure injection pump (3,000 gpm), and the operation of the core flooding tanks. The capability of other combinations of engineered safeguards to provide core protection has been evaluated in a preliminary analysis. This capability is shown on Figure 14-44-f.

In this evaluation the core is considered protected if the combination of emergency cooling systems considered will prevent core damage which would interfere with further core cooling. 1

The high pressure injection equipment with one pump operating 5
can accommodate leaks up to approximately 3 in. in diameter. The preliminary analysis upon which this conclusion is based indicates that this pump will probably have the capability to pro- 5
tect the core for leaks somewhat larger.

A combination of one high pressure pump and one low pressure 5
injection pump will protect the core up to a 0.4-ft^2 leak. This is equivalent to the rupture of a pressurizer surge line. One high pressure injection pump plus two low pressure injec- 5
tion pumps can protect the core up to leak sizes of 3.0 ft^2. This is considerably in excess of any of the piping connecting to the reactor coolant system. High pressure injection, plus the core flooding tanks and one low pressure injection pump, can protect the core up to 14.1 ft^2 which is a double-ended rupture of the 36-in. ID, hot leg piping.

The core flooding tanks and one low pressure injection pump can protect the core from about a 3-in. leak up to the 14.1-ft^2 leak. Figure 14-44-f demonstrates that high pressure injection system provides core protection for normal operating leakage and for small leaks in which pressure decay of the system may be slow. For intermediate leak sizes, either the core flooding tanks or low pressure injection protects the core following the loss-of-coolant accident. For very large leaks in the category of a double-ended rupture of the reactor coolant piping, the core flooding tanks and low pressure injection together protect the core. For these leaks the core flooding tanks provide immediate protection and can protect the core for several minutes following the rupture. Due to their limited volume, they must be supplemented by the high flow from the low pressure injection pumps within several minutes following the leak in order to prevent the core from again becoming uncovered as a result of boiling off the core flooding tank coolant.

This evaluation of emergency core cooling capability demonstrates that the core is protected for the entire spectrum of leak sizes in both hot and cold leg piping.

c. Reactor Building Design Base Accident 1

A range of leak sizes between 0.4 ft^2 and 14.1 ft^2 has been evaluated. The 14.1 ft^2 is equivalent to a double-ended rupture of the 36 in. ID reactor outlet piping. The reactor operating conditions used in this analysis are listed in Table 14-6.

The basis for this analysis is that only the makeup and purification system and the decay heat removal system are working. It was assumed that the makeup and purification system (high pressure injection) had one of the pumps available for operation and that the decay heat removal system 2
(low pressure injection) had both of the two pumps available for operation. These systems are assumed to operate on emergency power and can be in operation to deliver a total injection flow of 6,500 gpm within 24 sec after the accident occurs.

This approach is conservative since any combination of two flooding tank operations and minimum flow from the high and low pressure pumps will provide a lower energy release rate and peak reactor building pressures than those resulting from the 6,500 gpm flow. 2

During blowdown mass and energy releases to the reactor building are calculated by FLASH. Figure 14-43 is a plot of mass released to reactor building and Figure 14-44 is a plot of reactor coolant average pressure, each calculated by FLASH for the spectrum of hot leg ruptures. Following blowdown a 20-region SLUMP model was used to simulate the core thermal transient. This simulation includes fuel heat generation, metal-water reaction, and quenching when the injection water provided cooling by contact with the core.

As any given segment reached 4,800 F it was assumed to drop into water below the core and release all heat down to a datum of 281 F. Also, it was assumed that 10 per cent additional zirconium-water reaction occurred. When the water covered approximately 25 per cent of the core, the surface heat transfer coefficient from all the core clad to the water was assumed to be 100 Btu/hr-ft^2-F. The determination of water level was based on injection flow and included the effects of boiloff.

Assuming a pool boiling coefficient of 100 for the whole core when only 1/4 was covered was conservative for reactor building pressure analysis because it compressed overall energy transport into the shortest credible period.

Heat was also released from the hot metal of reactor coolant system and the reactor vessel internals. During the blowdown period a surface heat transfer coefficient of 1,000 Btu/hr-ft^2-F was used. After blowdown this coefficient was changed to 100 Btu/hr-ft^2-F for the metal below the leak and 5 Btu/hr-ft^2-F above the leak. The coolant sink temperature was provided by FLASH for the blowdown period and assumed to be 281 F thereafter. The internal heat transfer of the metal was based on a multilayer finite difference model. The whole process of reactor coolant system metal heat transfer was simulated with a digital computer program.

All heat transferred from the core and the reactor coolant system metal was assumed to generate steam without taking credit for the subcooled condition of the injection water (except for the portion which was boiled off) until the reactor vessel was filled to the leak height. Thereafter all energy was removed by low pressure injection flow of subcooled water, and the energy release to the reactor building atmosphere terminated. No delay was assumed in transporting steam to the reactor building. The heat from hydrogen burning was added directly to the reactor building as hydrogen was evolved from the metal-water reaction.

Both reactor inlet (cold) and reactor outlet (hot) line breaks were analyzed with FLASH. However, a complete analysis was made only for the hot line breaks since they provided for the most rapid heat transport from the core. This was true because the hot line breaks had longer blowdown and better heat transfer during blowdown than did the cold line breaks.

The results of calculations of fluid and heat transport to the reactor building as determined by FLASH, SLUMP, and other analytical models were used as input to the reactor building pressure analysis program, CONTEMPT.

Table 14-6

Reactor Operating Conditions for Evaluation

Parameter	Value
Reactor Coolant System Pressure, psig	2,185
Reactor Coolant Average Temperature, F	584
Reactor Power Level (ultimate), MWt	2,544
Reactor Coolant System Mass, lb	519,173
Initial Reactor Building Temperature, F	110
Initial Reactor Building Relative Humidity, %	0
Initial Reactor Building Pressure, psig	0

In calculating the reactor building pressure, it was assumed that the average temperature of the building atmosphere and structural materials was 110 F. Upon release of hot reactor coolant, the steel and concrete act as heat sinks which reduce the reactor building pressure. The heat sinks considered in this analysis are specified in Table 14-7. 1

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Table 14-7
Reactor Building Structure Data for Analysis of Time-Dependent Reactor Building Pressure

Parameter	Value
Reactor Building Free Volume, ft^3	2,000,000
Exposed Liner Plate	
Surface, ft^2	87,220
Mass, lb	1,238,000
Dome and Wall Liner Thickness, in.	0.375
Refueling Cavity Liner Thickness, in.	0.250
Reactor Building Concrete Enclosure Consisting of a 3-ft-Thick Dome and 3-ft, 6-in.-Thick Walls and a 2-ft-Thick Floor	
Wall and Dome Surface, ft^2	81,700
Wall and Dome Mass, lb	41,100,000
Exposed Floor Surface, ft^2	11,000
Exposed Floor Mass, lb	3,190,000
Structural and Miscellaneous Steel Exposed to Reactor Building Atmosphere	
Surface, ft^2	80,000
Mass, lb	500,000
Internal Concrete	
Surface, ft^2	102,280
Mass, lb	23,398,500
Refueling Cavity Concrete	
Surface, ft^2	5,520
Mass, lb	3,001,500

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Heat transfer from the reactor building atmosphere to the steel liner was calculated using a condensing coefficient of 620 Btu/hr-ft^2-F until a total heat input of 110 Btu/ft^2 had been achieved. Thereafter, a condensing coefficient of 40 Btu/hr-ft^2-F was used.

For heat transfer from the reactor building atmosphere to the concrete, a condensing coefficient of 40 Btu/hr-ft^2-F was used. For heat transfer from the sump water to the concrete floor a coefficient of 20 Btu/hr-ft^2-F was used. No credit was taken for heat transfer to reinforcing steel in the internal concrete structures.

For structural and miscellaneous steel, one heat transfer section with an equivalent thickness of 0.153 in. was used. Condensing coefficients of 620 and 40 Btu/hr-ft^2-F were used.

Following a loss-of-coolant accident, the reactor building is cooled by three reactor building emergency cooling units and a spray system. Each cooling arrangement has a heat removal capability of 240 x 10^6 Btu/hr at a vapor temperature of 281 F. Two cooling units plus 1,500 gpm sprays, or 3,000 gpm sprays, provide cooling that is at least equivalent to the three reactor building emergency cooling units. Each system is designed so that it alone can protect the reactor building against overpressure. Each system was assumed to operate on emergency power and was delayed until 35 sec after the rupture occurred.

Figure 14-45 shows the reactor building pressure for complete severance of a 36 in. ID reactor coolant system pipe (14.1 ft^2 rupture area) with 6,500 gpm of borated water injection into the reactor coolant system beginning 25 sec after the rupture. Reactor building cooling is provided by three emergency cooling units. The peak pressure resulting from this accident occurs 181 sec after the rupture at a value of 52.1 psig.

An analysis of the reactor building pressure for the 36 in. ID pipe rupture and spray cooling of the building has also been performed to demonstrate the effectiveness of this system. Initially coolant for the building sprays and for injection to the core is pumped from the borated water storage tank. When water from the borated water storage tank is depleted, the water collected in the reactor building sump is recirculated through the reactor building sprays and through the decay heat removal coolers to supply the low pressure injection water. The result is an increased injection and spray water temperature. No boiling of the injection water results from this decrease in subcooling. The reactor building spray effectiveness will decrease. The net result is a decrease in the energy removal rate from the reactor building atmosphere.

The requirements for cooling the water recirculated from the reactor building sump to the reactor building spray system are set by the design basis of this system. The design basis is to maintain the post-accident reactor building pressure below the design value. This criterion can be met by spraying the sump water directly into the reactor building atmosphere without additional cooling, other than that provided by the decay heat removal system.

0059

The water temperature in the reactor building sump during the recirculation phase of a loss-of-coolant accident is maintained below saturation temperature by the decay heat removal coolers. These coolers reduce the temperature of water recirculated to the reactor vessel and returned to the reactor building sump. The heat transfer surface of these coolers is set by the normal operating conditions under the decay heat removal operation mode. The cooling capability of this mode of operation will maintain the reactor coolant at 140 F or less at 20 hours after extended rated power operation and is in excess of that required under accident conditions. The performance of these coolers at various inlet temperatures is shown in Figure 6-4.

Figure 14-46 shows that the reactor building pressure decays to less than 5 psig in 24 hours. For comparison purposes and to show that the effect of spraying cooler water into the reactor building is small, a second curve is presented on Figure 14-46 which is based upon a spray recirculation cooling rate of 100 x 10^6 Btu/hr (approximately
equivalent to two decay heat removal coolers) at a sump temperature of 2
195 F. (This is the temperature of the sump when recirculation to the 3
sprays begins.) Figure 14-47 shows the temperature of the reactor building and sump coolant for the two conditions.

These curves demonstrate that cooling of the recirculated spray water has no effect on peak building pressure and only a minor effect on the rate of pressure decay during the first 24 hours. Accordingly, it is concluded that no cooling of the recirculated spray water is required for this accident.

Figures 14-48 through 14-52 show the reactor building pressure for the other rupture sizes analyzed with the same cooling capability as the 14.1 ft^2 rupture above. A summary of the input parameters and results for the spectrum analysis are tabulated in Table 14-8.

A 3.0 ft^2 rupture area results in the highest postaccident reactor building pressure (see Figure 14-49).

Figures 14-53 and 14-54 show the reactor building energy inventory as a function of time after rupture for 14.1 and 3 ft^2 rupture areas with three emergency coolers operating. These curves show the effectiveness of the reactor building structures and emergency cooling units.

Figures 14-55 and 14-56 show the reactor building vapor temperatures and sump temperatures following 14.1 and 3.0 ft^2 ruptures.

The peak reactor building pressure shown in this evaluation for the spectrum of leak sizes results is 52.1 psig and is the result of a 3.0 ft^2 rupture in the reactor outlet piping. The reactor building design pressure is 55 psig and a design margin of about 3 psi exists. With core flooding tank operation this margin would be increased further.

The above analyses conservatively assume that the hydrogen liberated will burn at the rate formed, and that no core flooding tank operation
occurs. The core flooding tanks limit the amount of zirconium water 3

reaction to 0.005 per cent for a 36 in. ID pipe rupture, and the potential hydrogen energy release is approximately 4,000 Btu. This amount of energy will not significantly affect reactor building pressure if ignition is delayed or if the hydrogen burns as formed.

For the case of no core flooding tanks, as used in the above reactor building design pressure evaluation, the amount of metal-water reaction is somewhat greater. The zirconium-water reaction begins at 40 sec and stops at 130 sec, by which time the 6,500 gpm of injection flow provides sufficient coolant inventory to the reactor vessel to recover the hot spot and quench the reaction. The steam flow during this period is assumed to provide the transport mechanism for the hydrogen generated. The resultant concentration of hydrogen (at time of maximum metal-water reaction rate) in the steam leaving the reactor vessel is 7.2 volume per cent. This concentration is below the flammability limit. Further dilution will occur as the steam enters the reactor building, and combustion will not occur, even as the reactor building is depressurized.

Criterion 17 of the AEC General Design Criteria states that the containment (reactor building) be designed to accommodate the largest credible energy release including the effects of credible metal-water reactions uninhibited by active quenching systems. Although the evaluation of the emergency injection systems demonstrates that only a small amount of metal-water reaction can occur, the case of no injection flow has been evaluated in response to the above criterion. This case assumed that, after blowdown, the reactor vessel would have water up to the bottom of the core. The core was allowed to heat up by decay heat and metal-water reaction heat.

Steam flow rate-limiting of the reaction was not considered so long as any water was assumed to be in the vessel. If and when the clad reached the melting temperature, it was assumed that the whole region slumped into the bottom of the vessel with the attendant reaction of 10 per cent more of the remaining zirconium and with the release to the reactor building of all sensible and latent heat above 281 F.

Upon completion of boiloff, heat input to the reactor building was assumed to cease. Figure 14-57 shows a reactor building pressure of 53.2 psig at 220 seconds, the time at which the reactor vessel boils dry. This peak pressure is below the 55 psig design pressure of the reactor building.

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Table 14-8
Summary of Reactor Building Pressure Analysis for
Reactor Building Emergency Cooling (240×10^6 Btu/hr)

Rupture Size, ft^2	14.1	8.5	3.0	2.0	1.0	0.4
Reference Figure No.	14-45	14-48	14-49	14-50	14-51	14-52
Time Blowdown Ends, sec	15	20	48	68	141	351
Time Low Pressure Injection Begins, sec	25	25	39	59	121	321
Fraction of Core Zr-reacted	0.08	0.05	< 0.01	zero	zero	zero
Time Zr-reaction Begins, sec	40	50	130	--	--	--
Time Zr-reaction Ends, sec	130	130	131	--	--	--
Time to Reach Peak Pressure, sec	181	181	41	67	181	261
Peak Building Pressure, psig	52.0	51.3	52.1	50.7	48.9	43.2
Vapor Temperature at Peak Pressure, F	278	277	278	276	274	266
Sump Temperature at Peak Pressure, F	232	230	221	215	210	196

Conditions for All Cases

a. 500 gpm high pressure injection

b. 6,000 gpm low pressure injection

c. Reactor hot leg rupture

d. No core flooding

e. No reactor building sprays

f. Three emergency cooling units start 35 sec after the rupture.

d. Reactor Building Zirconium Reaction Capability 1

In order to determine the theoretical ultimate zirconium reaction capability of the reactor building a series of hypothetical accidents was investigated.

Blowdown was based on the 14.1 ft^2 leak case. Heat transfer from the core and all reactor coolant system metal below the leak height was assumed to transfer to a 281 F sink based on a surface coefficient of 50,000 Btu/hr-ft^2-F. For reactor coolant system metal above the leak height 5 Btu/hr-ft^2-F was used.

Available core heat consisted of the initial stored heat, the equivalent of two full power seconds, decay heat, and metal-water reaction heat, which was added at arbitrary linear rates. The total heat transferred from the core and reactor coolant system metal was assumed to produce steam from water initially at the saturated condition. Hydrogen recombination energy was added to the reactor building as superheat at the rate of hydrogen production from the zirconium-water reaction.

A series of calculations for each of the various cooling capacities was made varying the energy input rate, i.e., Zr-H_2O reaction rate. For example, a 1 per cent per second zirconium-water reaction produces 1.173×10^6 Btu/sec of metal-water energy and 0.902×10^6 Btu/sec hydrogen recombination energy. In all cases the energy was input at a linear rate beginning 10 sec after the rupture. The emergency cooling units and spray coolers were started 35 sec after the rupture. The "time to complete reaction" is the time it takes to reach reactor building design pressure (55 psig).

The results of this study are presented in Figure 14-58. This amount of allowable zirconium reaction at any time after blowdown depends upon the amount of reactor building cooling in operation. The capability curves show that at approximately 10 sec, when the blowdown pressure peak occurs, the reactor building could accept an instantaneous zirconium-water reaction of 4 per cent. This capability increases greatly after the blowdown pressure peak with reactor building cooling equipment in operation.

With three emergency cooling units in operation a 100 per cent reaction in 4,200 sec will not exceed the design pressure of 55 psig. With three emergency cooling units and two sprays operating, a 100 per cent reaction in 1,420 seconds will not exceed the design pressure.

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14.2.2.3.5 Environmental Analysis of Loss-of-Coolant Accidents

Safety injection is designed to prevent significant clad melting in the event of a loss-of-coolant accident. The analyses in the preceding sections have demonstrated that safety injection will prevent clad melting for loss-of-coolant accidents resulting from reactor coolant system ruptures ranging in size from small leaks to the complete severance of a 36 in. ID main coolant pipe. Without clad melting, only the radioactive material in the coolant at the time of the accident plus some gap activity is released to the reactor building.

The environmental consequences from a loss-of-reactor-coolant accident are analyzed by assuming that 1 per cent of the fuel rods are defective before the release of reactor coolant to the reactor building. Table 11-3 lists the total activity in the coolant. In addition to the coolant activity, the activity associated with the gap of all fuel rods is also assumed to be released. Calculations indicate that 77 per cent of the fuel rods will have some point along their lengths with temperatures in excess of 1,200 F at the time of core flooding tank injection. While perforation of fuel cladding will require some time, it is conservatively assumed that all of the fuel rods release their gap activity during the accident.

Half of the iodine released is assumed to plate out on exposed surfaces in the reactor building. The other half is assumed to remain in the reactor building atmosphere where it is available for leakage. The sodium thiosulfate in the reactor building spray reduces the airborne iodine as described below. Of the iodine available for leakage, 5 per cent has been conservatively assumed to be unavailable for removal by the spray.

The rate at which the elemental iodine can be removed from the reactor building atmosphere by the reactive spray is calculated using Griffith's methods.[14] This method is based on the work of Taylor,[15] who showed that the rate at which elemental iodine can be transferred into reactive solutions is controlled by the gas film resistance, and on the work of Ranz and Marshall,[16] who showed that the equation below can be used to calculate the mass transfer coefficient when the rate of transfer is controlled by the gas film resistance:

$$k_G = \frac{D\rho M_I}{M_m dP}\left[2 + 0.6\left(\frac{dv\rho}{\mu}\right)^{1/2}\left(\frac{\mu}{D\rho}\right)^{1/3}\right]$$

where

k_G = gas film mass transfer coefficient, gm/cm^2-sec-atmos

D = diffusivity of iodine in air, cm^2/sec

ρ = density of air, gm/cm^3

M_I = molecular weight of iodine, gm/gm-mole

M_m = mean molecular weight of the air-iodine mixture in the boundary layer

P = partial pressure of air in the gas film, atmos

d = drop diameter, cm

v = relative velocity between the drop and the gas phase, or approximately the terminal velocity of the drop, cm/sec

μ = viscosity of the air

Since the mass transfer of iodine is gas-film-controlled, k_G is approximately equal to K_G (below), and the foregoing equation can be rewritten in terms of the velocity of deposition, V_g:

$$V_g = \frac{RT}{M_I} K_G = \frac{RT}{M_I} \frac{DM_I}{M_m d} \frac{\rho}{P} \left[2 + 0.6 \left(\frac{dv\rho}{\mu} \right)^{1/2} \left(\frac{\mu}{D\rho} \right)^{1/3} \right]$$

where

V_g = overall velocity of deposition, cm/sec

R = universal gas constant = 82.057, atmos-cm^3/K-gm-mole

T = absolute temperature, K

K_G = overall mass transfer coefficient, gm/cm^2-sec-atmos

Since the maximum possible iodine concentration in the large volume in the reactor building is less than 10^{-7} gm/cc, the partial pressure of air in the gas film, P, can be taken as the total pressure, and the mean molecular weight, M_m, can be taken as the molecular weight of air, M_A. If the gas equation is used, the equation may be simplified somewhat by substituting M_A/RT for ρ/P, as follows:

$$V_g = \frac{D}{d} \left[2 + 0.6 \left(\frac{dv\rho}{\mu} \right)^{1/2} \left(\frac{\mu}{D\rho} \right)^{1/3} \right]$$

The surface area of drops available for iodine absorption can be calculated from the next equation, which is based on the assumption that all the drops are spherical and have the same diameter.

$$S = \frac{\pi d^2 F \theta}{\frac{\pi}{6} d^3} = \frac{6F\theta}{d} = \frac{6FH}{dv}$$

where

S = surface area of drops suspended in the gas phase, cm^2
F = spray flow rate, cm^3/sec
θ = drop fall time, sec
d = drop diameter, cm
H = drop fall height, cm
v = drop fall velocity or terminal velocity, cm/sec

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If there is a large excess of chemical reagent to react with the iodine and convert it to a nonvolatile form with little or no tendency to return to the gas phase, then the iodine removal rate can be expressed by

$$\frac{dI}{dt} = -\left(\frac{V_g S}{V_c}\right) I = -\lambda_s I$$

where

V_c = free volume of reactor building, cm^3 or ft^3

λ_s = iodine removal time constant, hr^{-1}

The fraction remaining in the reactor building atmosphere is expressed as a function of time by the solution of the equation above as follows:

$$\frac{I}{I_o} = e^{-\lambda_s t}$$

where

$\frac{I}{I_o}$ = fraction of initial inventory remaining

t = spray time, hr

When the specific parameters for Unit 3 or 4 of the Crystal River Plant are used:

F = 3,000 gpm	v = 397 cm/sec
H = 90 ft	V_g = 5.06 cm/sec
$V_c = 2 \times 10^6$ ft^3	$\lambda_s = \frac{6V_g FH}{V_c dv} = 25.3\ hr^{-1}$
d = 1,000 microns	

These iodine removal calculations have conservatively corrected the iodine deposition velocity (V_g) to the peak temperature and pressure in the reactor building. A sensitivity analysis was performed on the iodine removal calculations, and the results are shown in 14.2.2.4.3 in terms of the 2-hour iodine dose at the exclusion distance following an MHA.

Although the reactor building leakage rate will decrease as the pressure decays, the leakage is assumed to remain constant at the rate of 0.25 per cent per day for the first 24 hours. Thereafter, since the reactor building will have returned to nearly atmosphere pressure, the rate is assumed to be reduced to 0.125 per cent per day and remain at this value for the duration of the accident.

The atmospheric dispersion characteristics of the Plant site are described in 2.3. The site dispersion factors for the duration of the accident are listed in Table 2-3. A breathing rate of 3.47×10^{-4} m^3/sec is assumed for the 2-hour exposure. For the 24-hour exposure, a breathing rate of 3.47×10^{-4} m^3/sec is assumed for the first 8 hours, and a rate of 1.74×10^{-4} m^3/sec is assumed for the remaining 16 hours. For the 30-day exposure, a breathing rate of 2.32×10^{-4} m^3/sec is assumed.

The iodine doses to the thyroid per curie inhaled are obtained from the values given in TID-14844:

I-131	1.48×10^6 rem per curie
I-132	5.35×10^4 rem per curie
I-133	4.0×10^5 rem per curie
I-134	2.5×10^4 rem per curie
I-135	1.24×10^5 rem per curie

Figure 14-59 shows the total integrated dose to the thyroid as a function of distance from the reactor building for 2-hour, 24-hour, and 30-day exposures. The total thyroid dose at the 4,400 ft exclusion distance is 1.45 rem for a 2-hour exposure, 8.0 rem for a 24-hour exposure, and 9.9 rem for a 30-day exposure. These doses are well below the guideline values of 10 CFR 100. The direct dose from this accident is insignificant since it is several orders of magnitude below 10 CFR 100. | 1

14.2.2.3.6 Effects of Reactor Building Purging

At times during the normal operation of the reactor, it may be desirable to purge the reactor building while the reactor is operating. In the event a loss-of-coolant accident were to occur during purging operations, activity would be released to the environment. The purge values will be completely closed in 5 sec. During this time, assuming a 36 in. ID, double-ended rupture, essentially all of the reactor coolant will have been blown down. The activity in the reactor building is due to the reactor coolant activity after operation with 1 per cent failed fuel. For this case, 0.53 per cent of the reactor building atmosphere will escape through the purge valves before they close, corresponding to a release of 3 equivalent curies of iodine-131. This analysis assumes unrestricted flow through the purge line for the full 5-second closing time. No reduction in flow is assumed as the valve closes, and therefore the results are conservative. The release of this iodine results in a total integrated thyroid dose of 0.48 rem at the exclusion distance. This dose, when added to the thyroid dose for a loss-of-coolant accident without purging, is well below the 10 CFR 100 guidelines. Therefore, purging operations can be performed during reactor c ·tion.

14.2.2.4 Maximum Hypothetical Accident

14.2.2.4.1 Identification of Accident

The analyses in the preceding sections have demonstrated that even in the event of a loss-of-coolant accident, no significant core melting will occur. However, to demonstrate that the operation of a nuclear power plant at the proposed site does not present any undue hazard to the general public, a hypothetical accident involving a gross release of fission products is evaluated. No mechanism whereby such a release occurs is postulated since it would require a multitude of failures in the engineered safeguards provided to prevent its occurrence. Fission products are assumed to be released from the core as stated in TID-14844, namely, 100 per cent of the noble gases, 50 per cent of the halogens, and 1 per cent of the solids.

Further, 50 per cent of the iodines released to the reactor building are assumed to plate out. Other parameters, such as meteorological conditions, iodine inventory of the fuel, reactor building leak rate, reactor building iodine removal rate, etc., are the same as those assumed for the loss-of-coolant accident in 14.2.2.3.5. The average iodine inventory, in terms of equivalent curies of iodine-131 available for leakage at different time periods after the accident, is as follows:

0 to 2 hours	28.7×10^6 curies
0 to 24 hours	22.8×10^6 curies
1 to 30 days	5.1×10^6 curies

14.2.2.4.2 Analysis and Results of Environmental Analysis

Figure 14-60 presents the total integrated dose to the thyroid as a function of distance from the reactor building for 2-hour, 24-hour, and 30-day exposures. It can be seen that the 2-hour thyroid dose of 65 rem at the exclusion distance of 4,400 ft and the 30-day thyroid dose of 3.4 rem at the 41-mile low population zone distance are less than the guideline values of 10 CFR 100. In the year 2015, the projected population within a 5-mile radius of the Plant will be less than 1,000. The corresponding 30-day thyroid dose from the MHA at the 5-mile zone boundary is 38 rem.

The direct dose to the whole body following the accident is shown in Figure 14-61. No significant dose exists from this source at the exclusion distance.

The dose to the whole body from the passing cloud has been calculated using the same meteorological conditions used for determining the thyroid dose. The 2-hour whole body dose at the exclusion distance is only 1.9 rem, and the 30-day dose at the 41-mile low population zone distance is 0.11 rem. The 30-day dose at the 5-mile zone boundary is 1.2 rem.

14.2.2.4.3 Effects of a Sensitivity Analysis of the Reactor Building Sprays for Iodine Removal

A sensitivity analysis on the calculation of iodine removal was performed using the reactive chemical sprays in the reactor building. The results are shown in Table 14-9 in terms of the 2-hour iodine dose at the exclusion distance following an MHA.

Table 14-9

Sensitivity Analysis Showing the Effect of Parameters on the Two-Hour Iodine Dose at the Exclusion Distance Following an MHA

Case No.	Drop Size, microns	Drop Fall Velocity, cm/sec	Velocity of Deposition, cm/sec	Temp, F	Press., psig	Iodine(1) Removal Time Constant, hr^{-1}	Iodine(1) Dose, rem	Iodine(2) Removal Time Constant, hr^{-1}	Iodine(2) Dose, rem	Remarks
1	1,000	397	5.06	281	55	12.65	82	25.3	65	Operation of the reactor building spray system at maximum building temperature and pressure.
2	1,000	397	6.44	212	25	16.05	75	32.1	51	Operation of the reactor building spray system after partial cooling, about 1 hour.
3	1,000	397	11.55	100	0	28.8	62	57.6	54	Operation of the reactor building spray system after cooling to ambient conditions.
4	1,000	3,970	14.83	281	55	3.7	171	7.39	109	Effect of drop falling at 10 times its terminal velocity.
5	2,000	649	4.25	281	55	3.25	192	6.50	116	Effect of large drop size.
6	200	76	7.24	281	55	471	45	942	45	Effect of small drop size.

For all cases, reactor building free volume = 2×10^6 ft^3, and drop fall height = 90 ft.

Notes: (1) Flow rate of sprays = 1,500 gpm.

(2) Flow rate of sprays = 3,000 gpm.

14.2.2.4.4 Effects of Engineered Safeguards Leakage During the Maximum Hypothetical Accident

An additional source of fission product leakage during the maximum hypothetical accident can occur from leakage of the engineered safeguards external to the reactor building during the recirculation phase for long-term core cooling. A detailed analysis of the potential leakage from these systems is presented in 6.3. That analysis demonstrated that the maximum leakage is about 5,000 cc/hr.

It is assumed that the water being recirculated from the reactor building sump through the external system piping contains 50 per cent of the core saturation iodine inventory. This is the entire amount of iodine release from the reactor cooling system. The 50 per cent escaping from the reactor coolant system is consistent with TID-14844. The assumption that all the iodine escaping from the reactor coolant system is absorbed by the water in the reactor building is conservative since much of the iodine released from the fuel will be plated out on the building walls. The activity in the recirculation water is equal to 0.037 equivalent curies of I-131 per cc of water. The iodine is chemically bound to the sodium thiosulfate, and will not be released to the atmosphere. However, it is conservatively assumed that iodine release does occur. Since the temperature of water in the reactor building sump is less than 200 F when recirculation occurs, the iodine release is calculated using a gas/liquid partition coefficient of 9×10^{-3}.

Leakage from the auxiliary building is caused by exfiltration. The most restrictive case for a ground release occurs during inversion conditions. It is assumed that the building leaks at the rate of 100 per cent per day with atmospheric dilution occurring in the wake of the building. For this building leak rate and the inversion condition, the iodine will produce an integrated dose to the thyroid of 0.005 rem in 2 hours at the 4,400 ft exclusion distance.

0070

14.3 REFERENCES

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(3) Dispersion of Soluble Radioactive Material in Water, CF-58-3-109.

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(6) Ackerman, R., et al., "High Temperature Vapor Pressure of UO_2", Journal of Chemical Physics, December 1956.

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(8) AEC Research and Development Reports, WIGL2 - A Program for the Solution of the One-Dimensional Two-Group, Space-Time Diffusion Equations Accounting for Temperature, Xenon and Control Feedback, WAPD-TM-532, October 1965.

(9) Margolis, S. G. and Redfield, J. A., FLASH: A Program for Digital Simulation of the Loss-of-Coolant Accident, WAPD-TM-534, May 1966.

(10) Grenda, R. J. and Patterson, J. F., "The Velocity of Rising Steam in a Bubbling Two-Phase Mixture", Transactions of The ANS 5, No. 1, p 151, June 1962.

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(12) Wagner, R. J. and Finnegan, L. J., "An Analytical Model for Predicting the Pressure-Temperature History Within a Containment Vessel in Response to a Loss-of-Coolant Accident", Phillips Petroleum Company, Atomic Energy Division, Idaho Falls, Idaho, Presented at ANS Meeting, Washington, D.C., November 1965.

(13) Quinn, E. P., Forced-Flow Heat Transfer to High-Pressure Water Beyond the Critical Heat Flux, ASME 66WA/HT-36, November 27, 1966.

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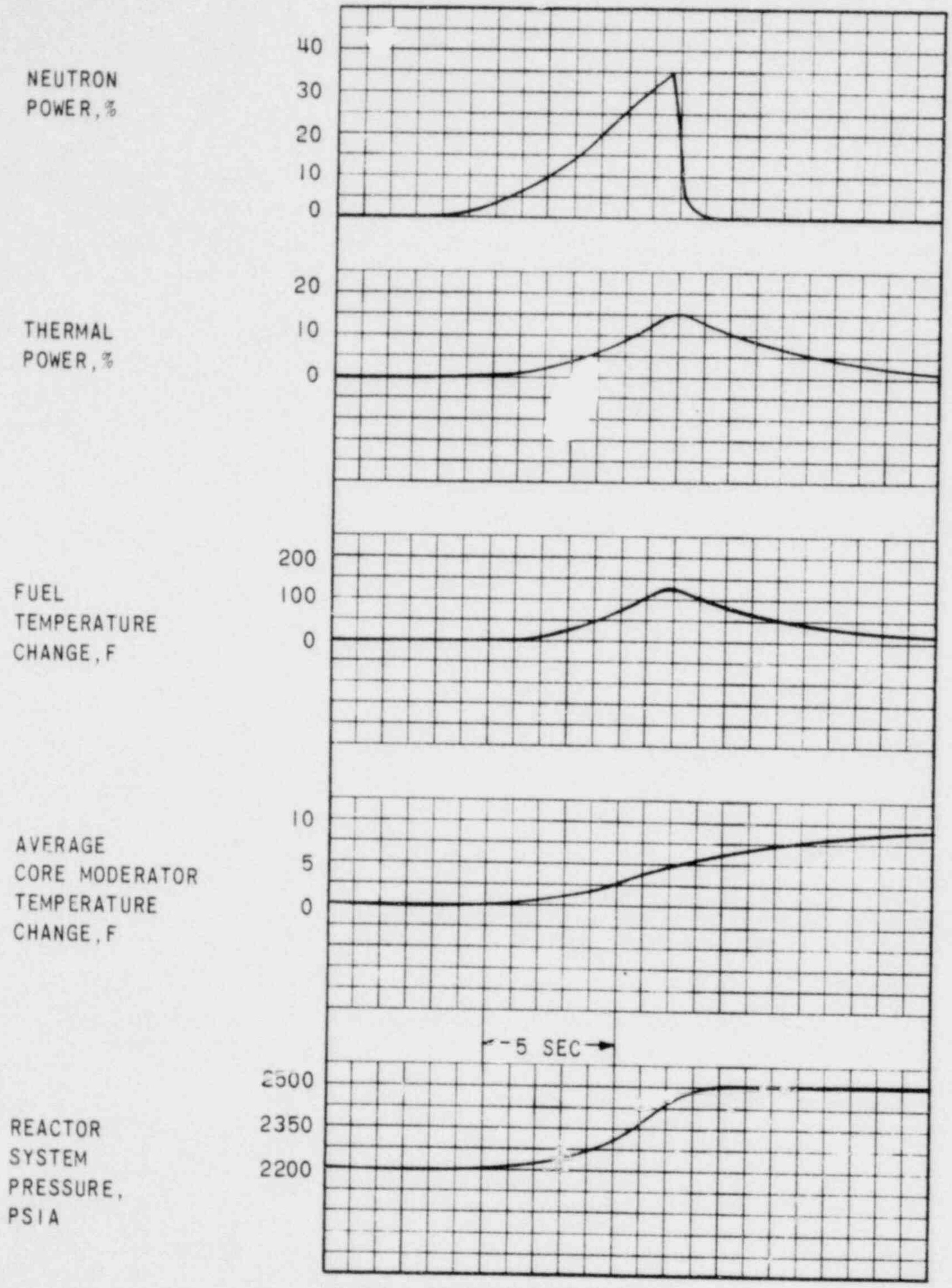
(16) Ranz, W. E. and Marshall, W. R., Chem. Eng. Progress, 48, 141, 173, 1952.

0071

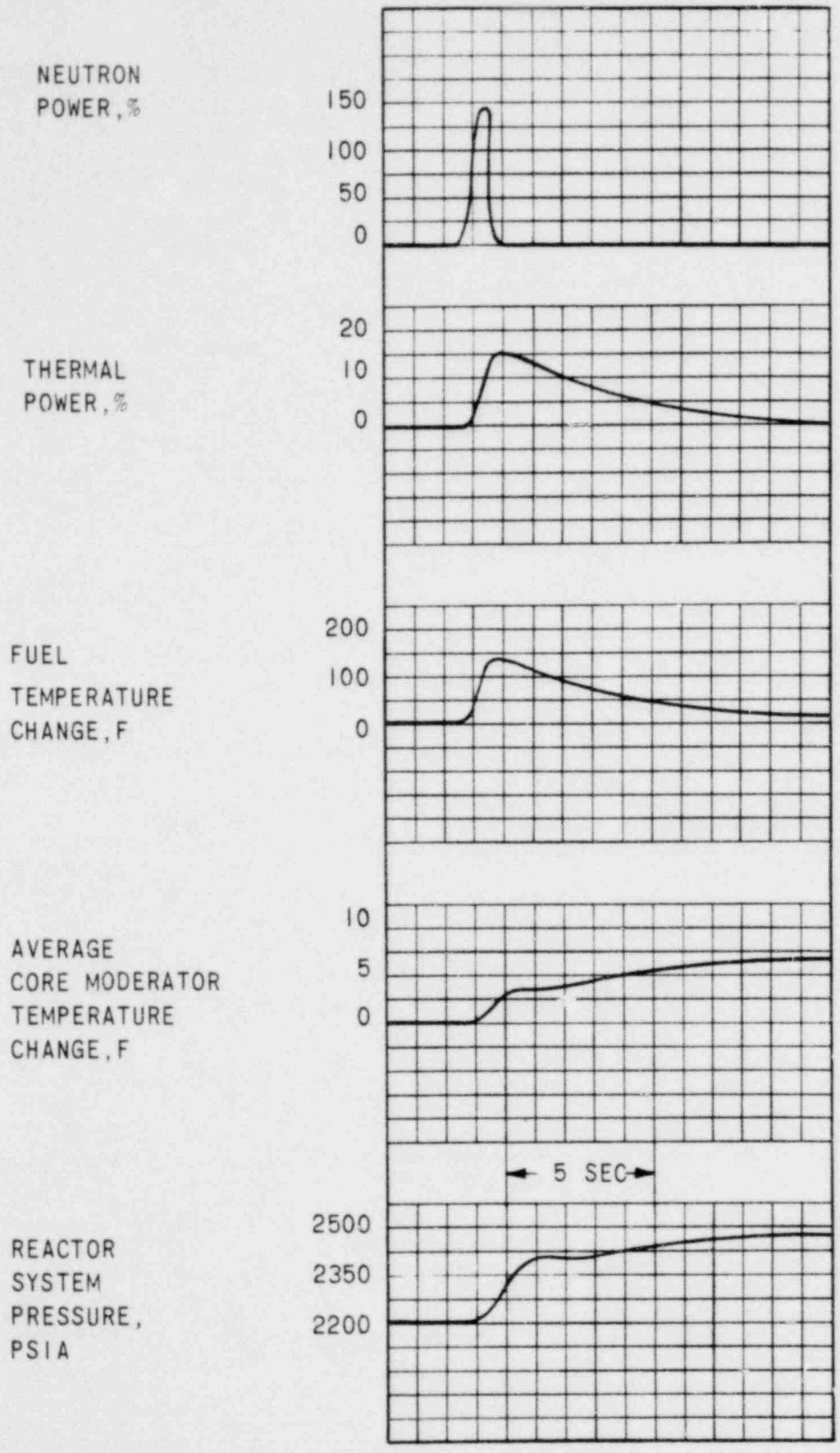


START-UP ACCIDENT FROM 10^{-9} RATED POWER USING A 1.2% ΔK/K ROD GROUP; HIGH PRESSURE REACTOR TRIP IS ACTUATED

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-1



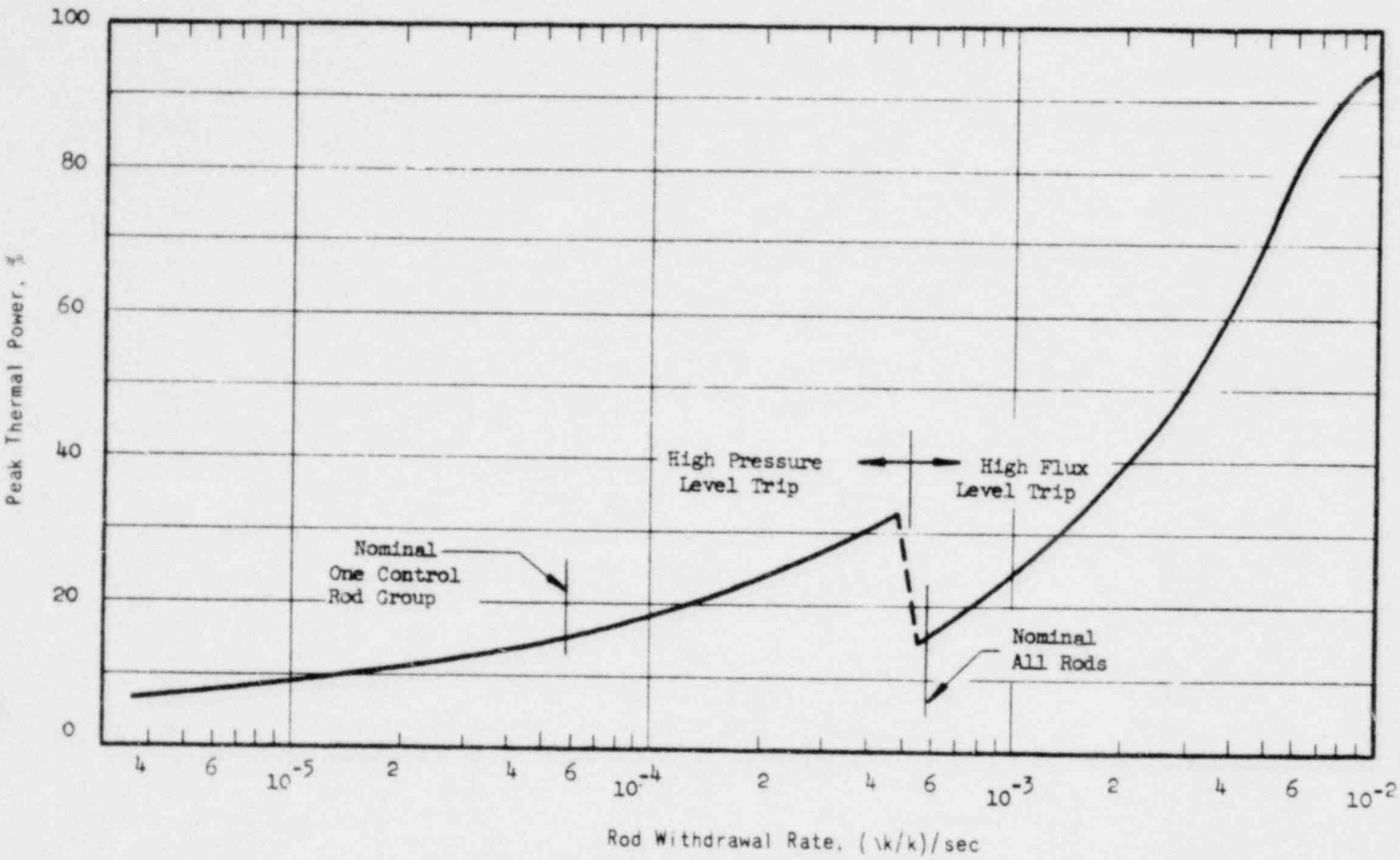
START-UP ACCIDENT FROM 10^{-9} RATED POWER USING ALL RODS WITH A WORTH OF 9.5% $\Delta K/K$; HIGH FLUX REACTOR TRIP IS ACTUATED

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-2

0073

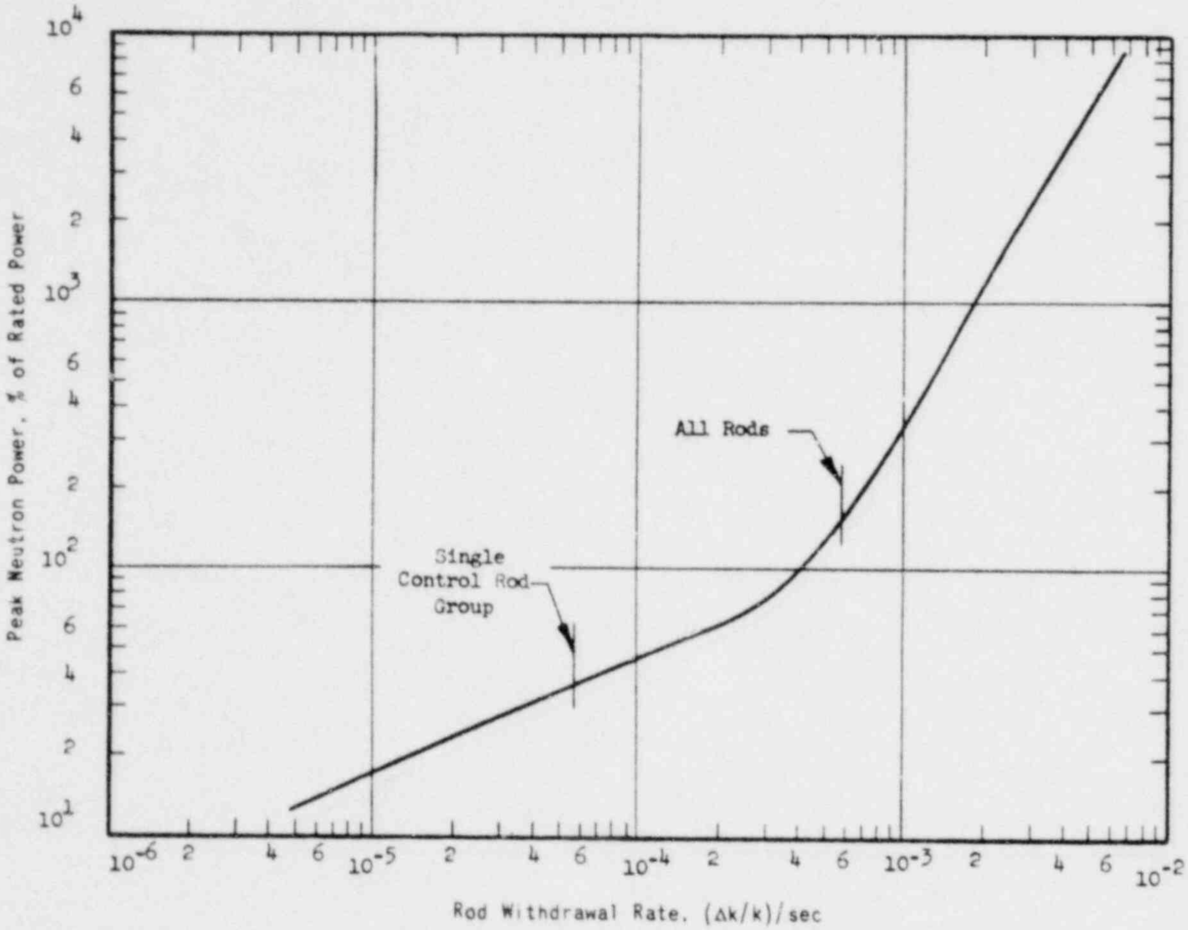


PEAK THERMAL POWER VERSUS ROD WITHDRAWAL RATE FOR A START-UP ACCIDENT FROM 10^{-9} RATED POWER

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-3

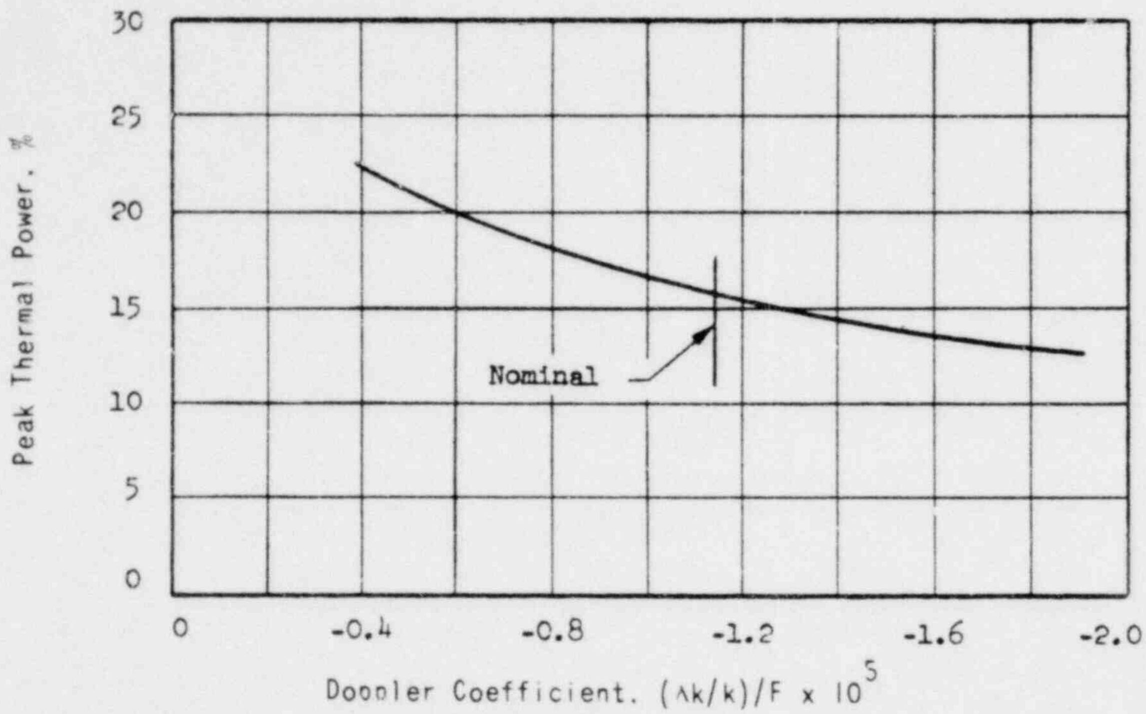


PEAK NEUTRON POWER VERSUS ROD WITHDRAWAL RATE FOR A START-UP ACCIDENT FROM 10^{-9} RATED POWER

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-4



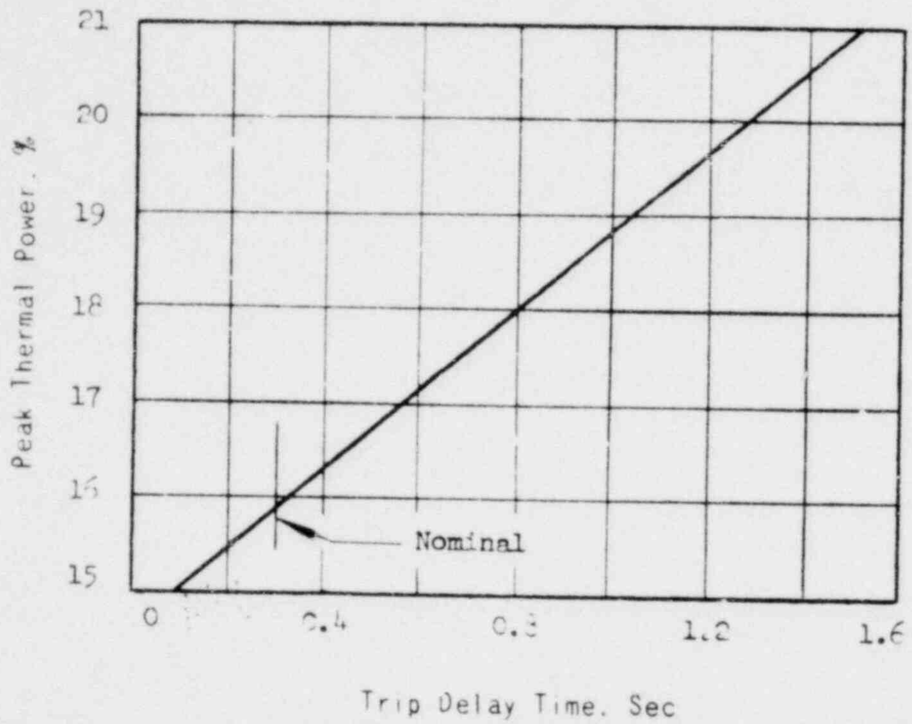
PEAK THERMAL POWER VERSUS DOPPLER COEFFICIENT FOR A START-UP ACCIDENT USING A 1.2% $\Delta K/K$ ROD GROUP AT 5.8×10^{-5} $(\Delta K/K)$/SEC. FROM 10^{-9} RATED POWER

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-6

0077

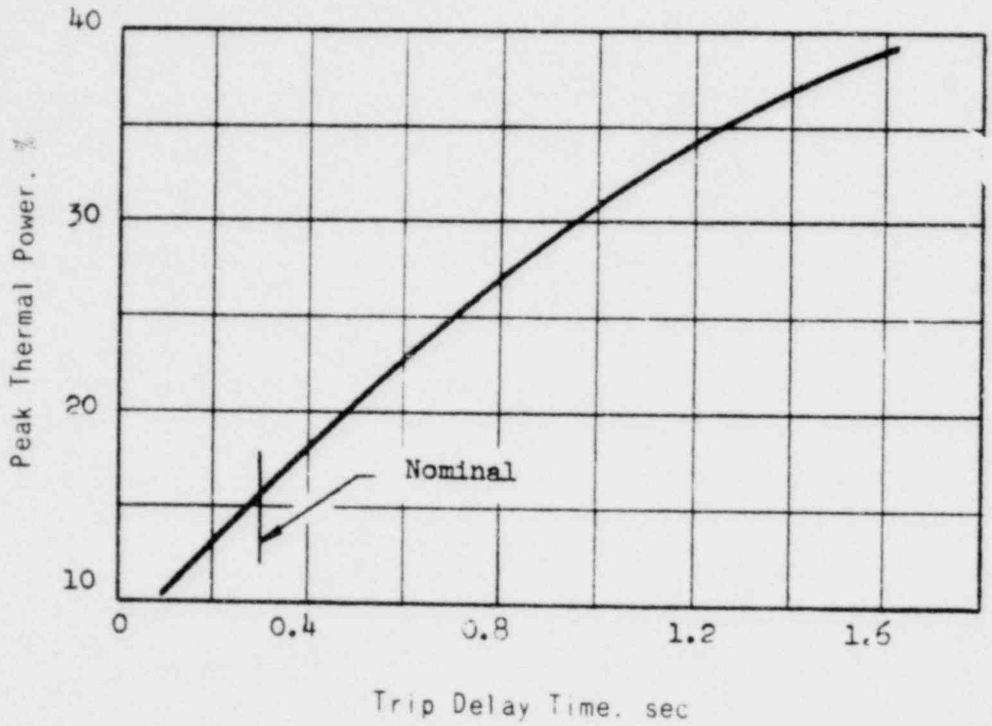


PEAK THERMAL POWER VERSUS TRIP DELAY TIME FOR A START-UP ACCIDENT USING A 1.2% $\Delta K/K$ ROD GROUP AT 5.8×10^{-5} $(\Delta K/K)$/SEC. FROM 10^{-9} RATED POWER

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-5

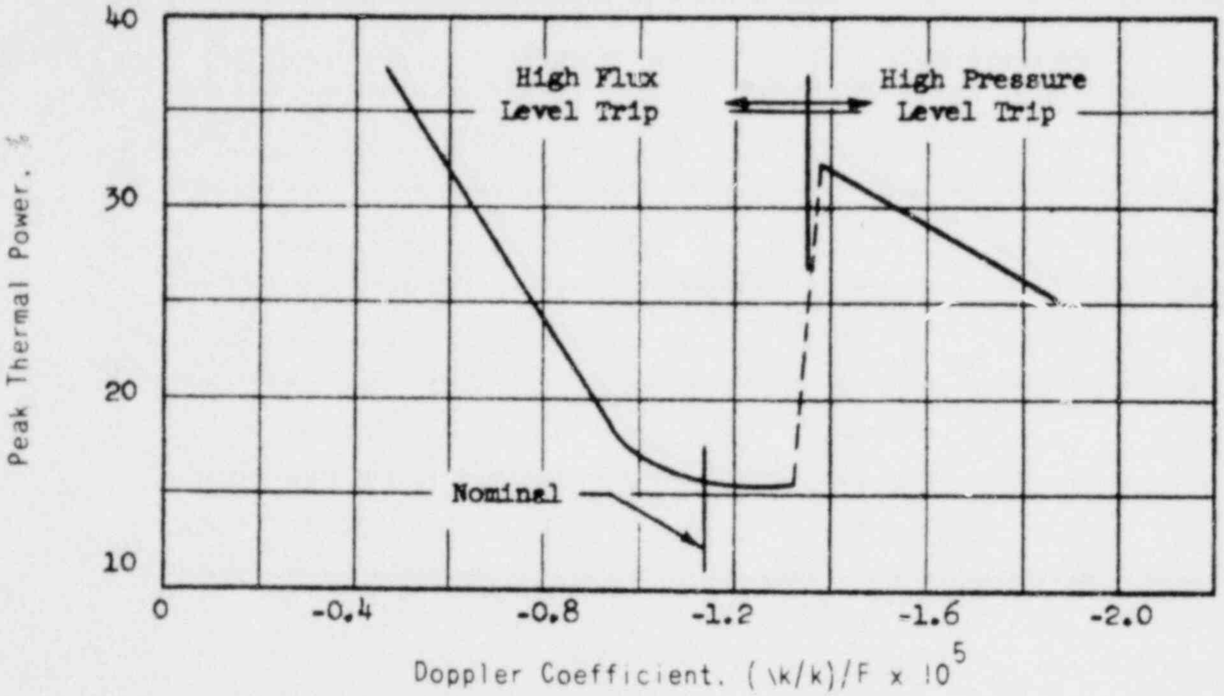


PEAK THERMAL POWER VERSUS TRIP DELAY TIME FOR A START-UP ACCIDENT USING ALL RODS AT 5.8×10^{-4} ($\Delta K/K$)/SEC. FROM 10^{-9} RATED POWER

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-7



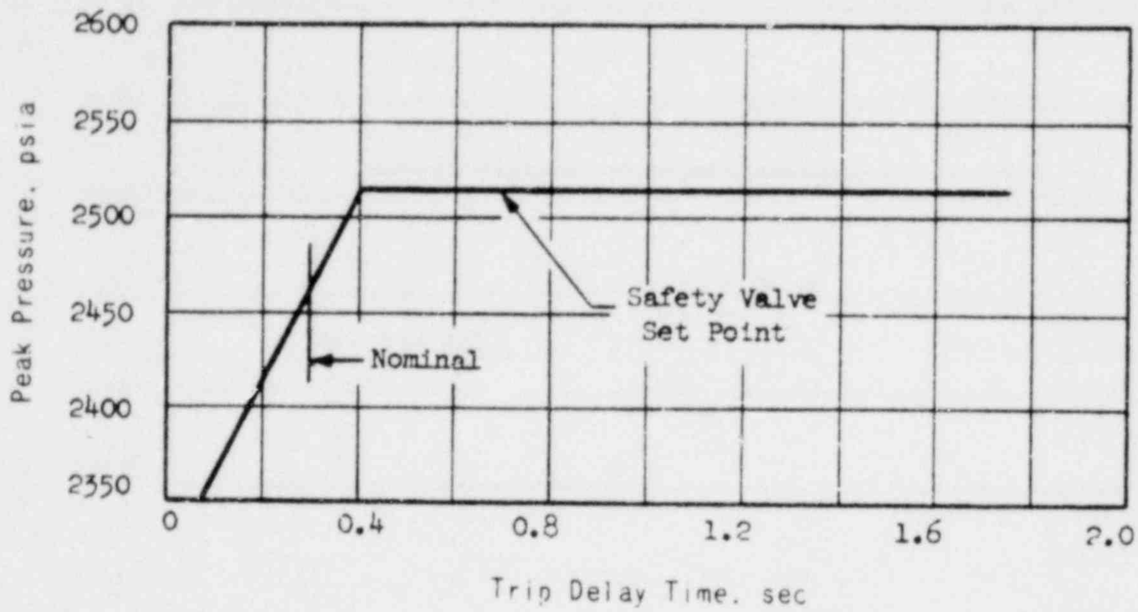
PEAK THERMAL POWER VERSUS DOPPLER COEFFICIENT FOR A START-UP ACCIDENT USING ALL RODS AT 5.8×10^{-4} ($\Delta K/K$)/SEC. FROM 10^{-9} RATED POWER

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-8

0079

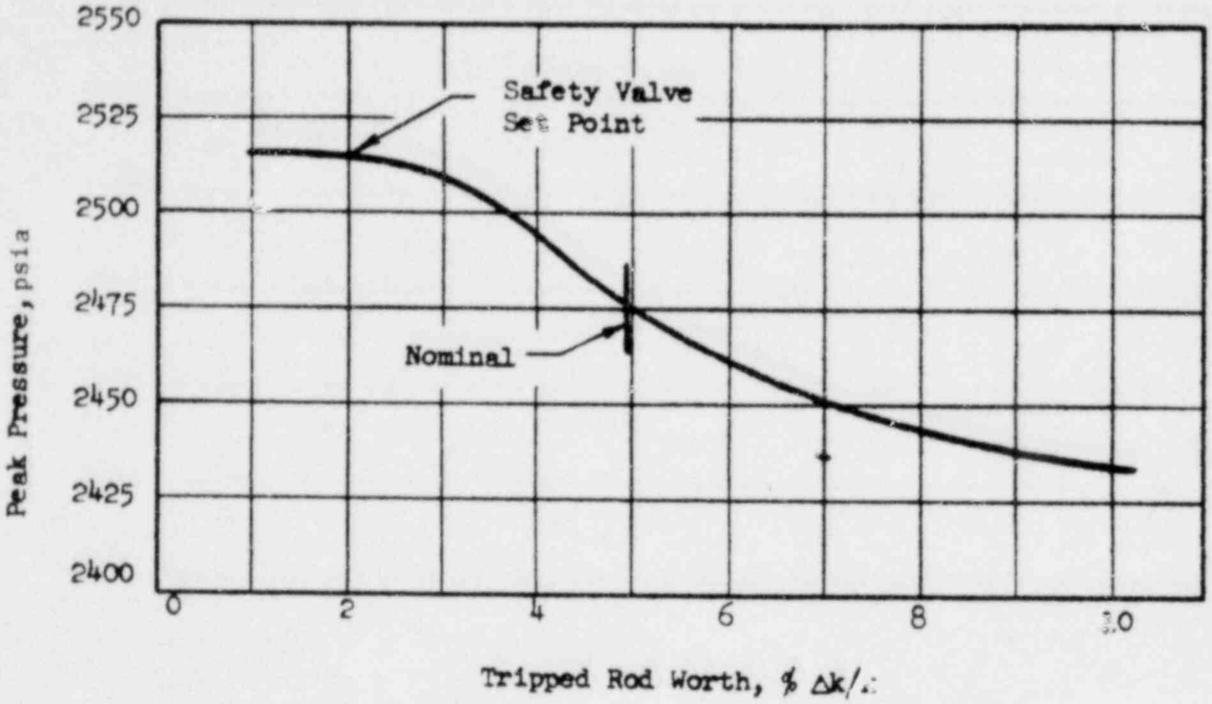


PEAK PRESSURE VERSUS TRIP DELAY TIME FOR A START-UP ACCIDENT USING ALL RODS AT 5.8×10^{-4} $(\Delta K/K)$/SEC. FROM 10^{-9} RATED POWER

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-9



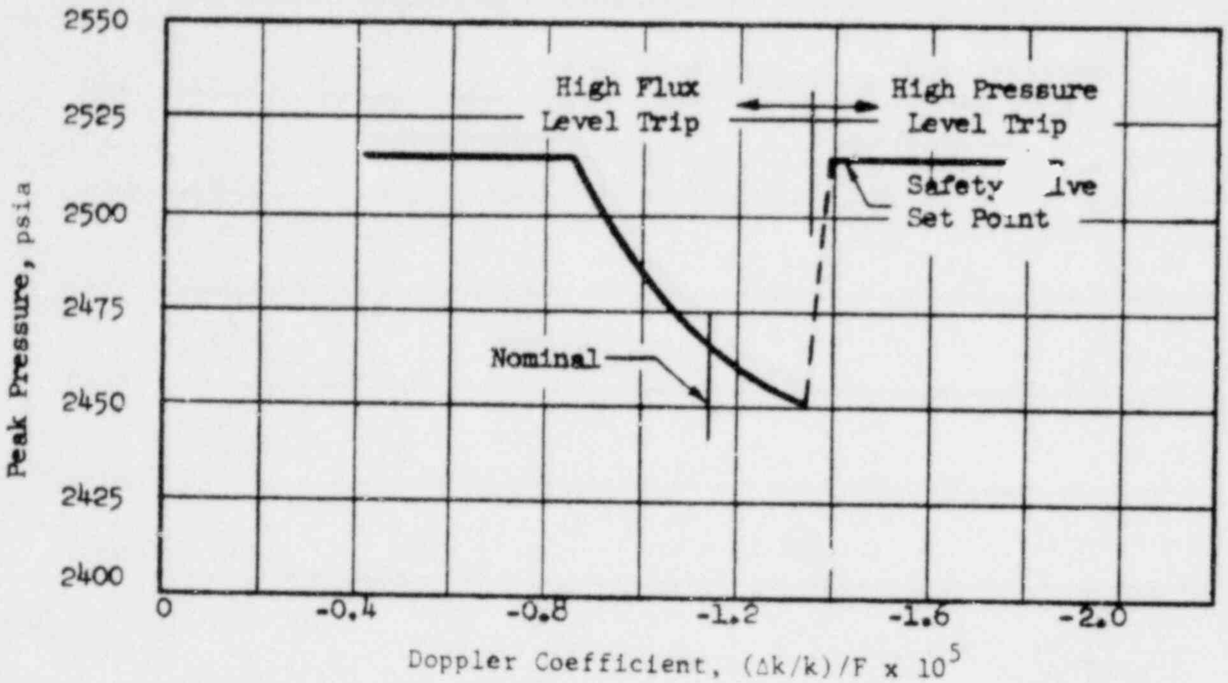
PEAK PRESSURE VERSUS TRIPPED ROD WORTH FOR
A START-UP ACCIDENT USING ALL RODS AT
5.8×10^{-4} (ΔK/K)/SEC. FROM 10^{-9} RATED POWER

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-10

0081

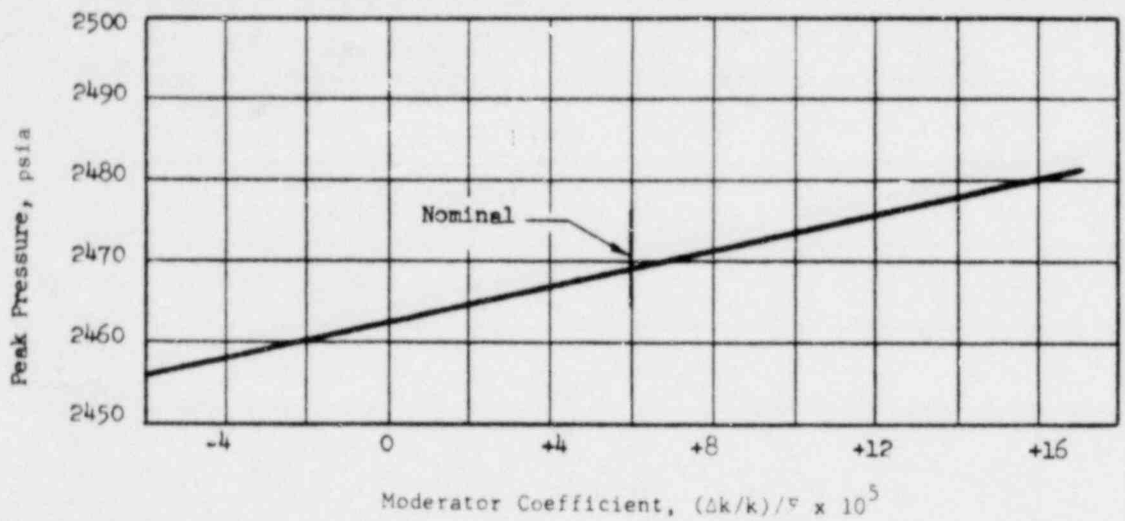


0082

PEAK PRESSURE VERSUS DOPPLER COEFFICIENT FOR A START-UP ACCIDENT USING ALL RODS AT 5.8×10^{-4} $(\Delta K/K)$/SEC. FROM 10^{-9} RATED POWER

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION FIGURE 14-11



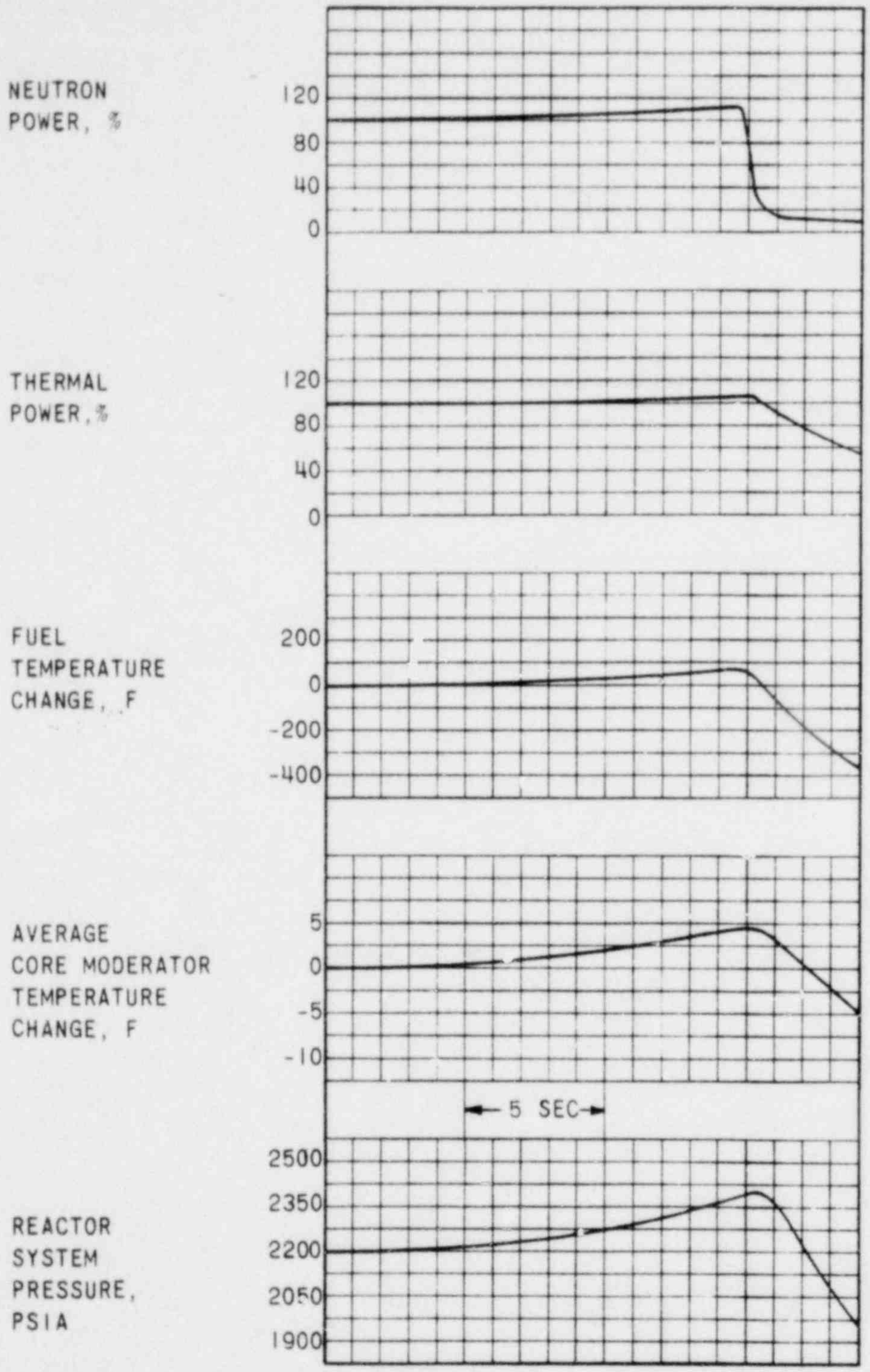
PEAK PRESSURE VERSUS MODERATOR COEFFICIENT FOR A START-UP ACCIDENT USING ALL RODS AT 5.8×10^{-4} ($\Delta K/K$)/SEC. FROM 10^{-9} RATED POWER

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-12

0083



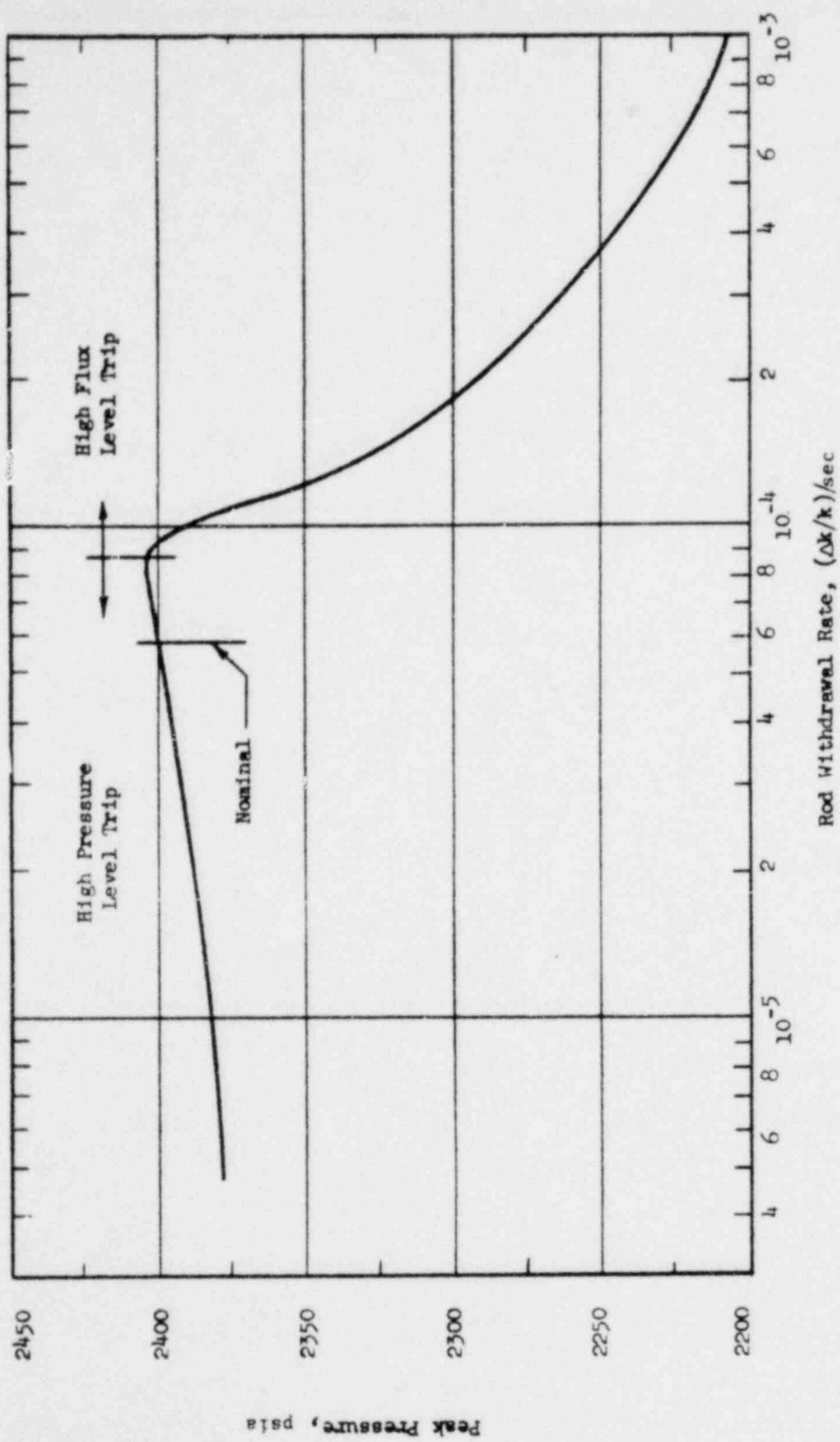
0084

ROD WITHDRAWAL ACCIDENT FROM RATED POWER USING A 1.2% ΔK/K ROD GROUP AT 5.8×10^{-5} (ΔK/K)/SEC; HIGH PRESSURE REACTOR TRIP IS ACTUATED

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

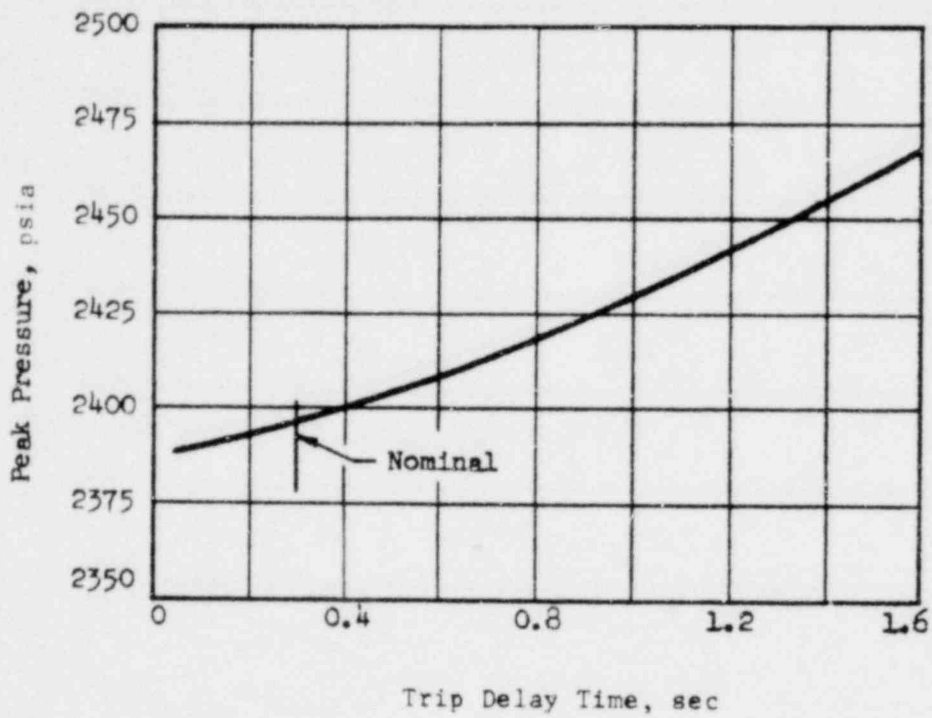
FIGURE 14-13



PEAK PRESSURE VERSUS ROD WITHDRAWAL RAT…
FOR A ROD WITHDRAWAL ACCIDENT
FROM RATED POWER

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION FIGURE 14-14

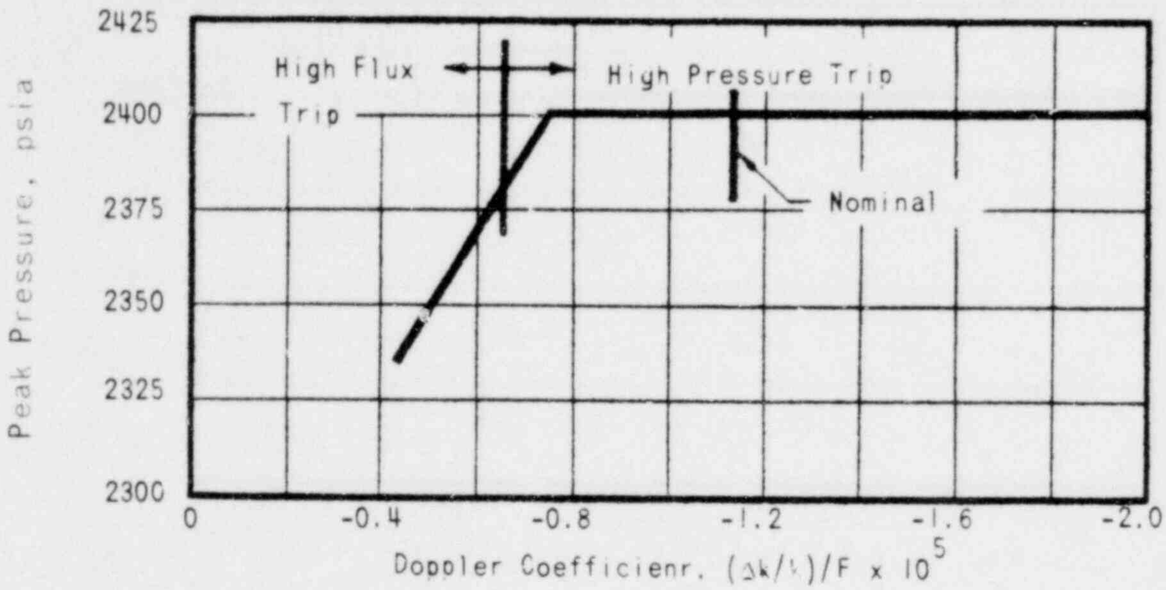


PEAK PRESSURE VERSUS TRIP DELAY TIME FOR A ROD WITHDRAWAL ACCIDENT FROM RATED POWER USING A 1.2% ΔK/K ROD GROUP; HIGH PRESSURE REACTOR TRIP IS ACTUATED

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-15



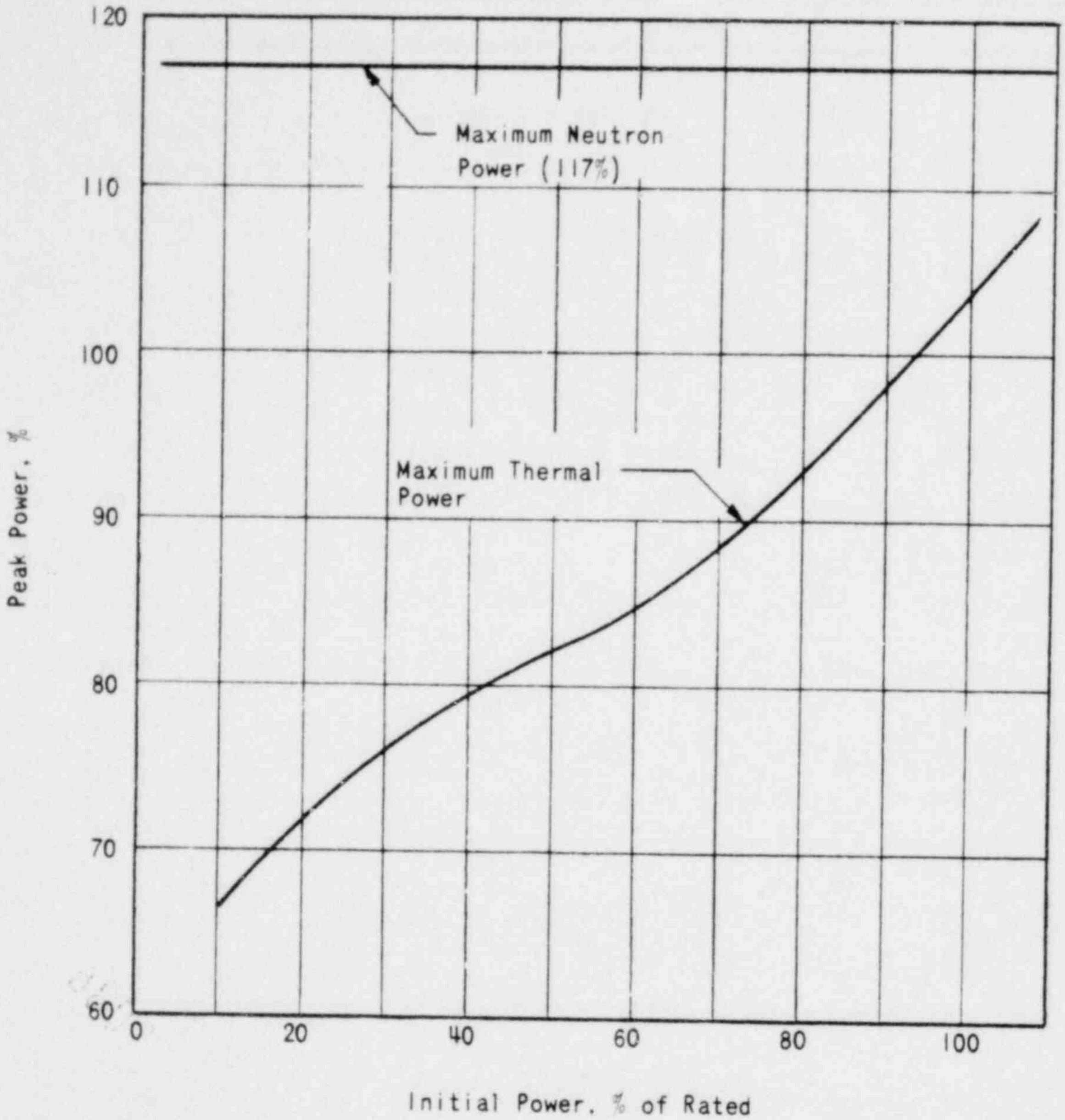
0087

PEAK PRESSURE VERSUS DOPPLER COEFFICIENT FOR A ROD WITHDRAWAL ACCIDENT FROM RATED POWER USING A 1.2% ΔK/K ROD GROUP

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-16



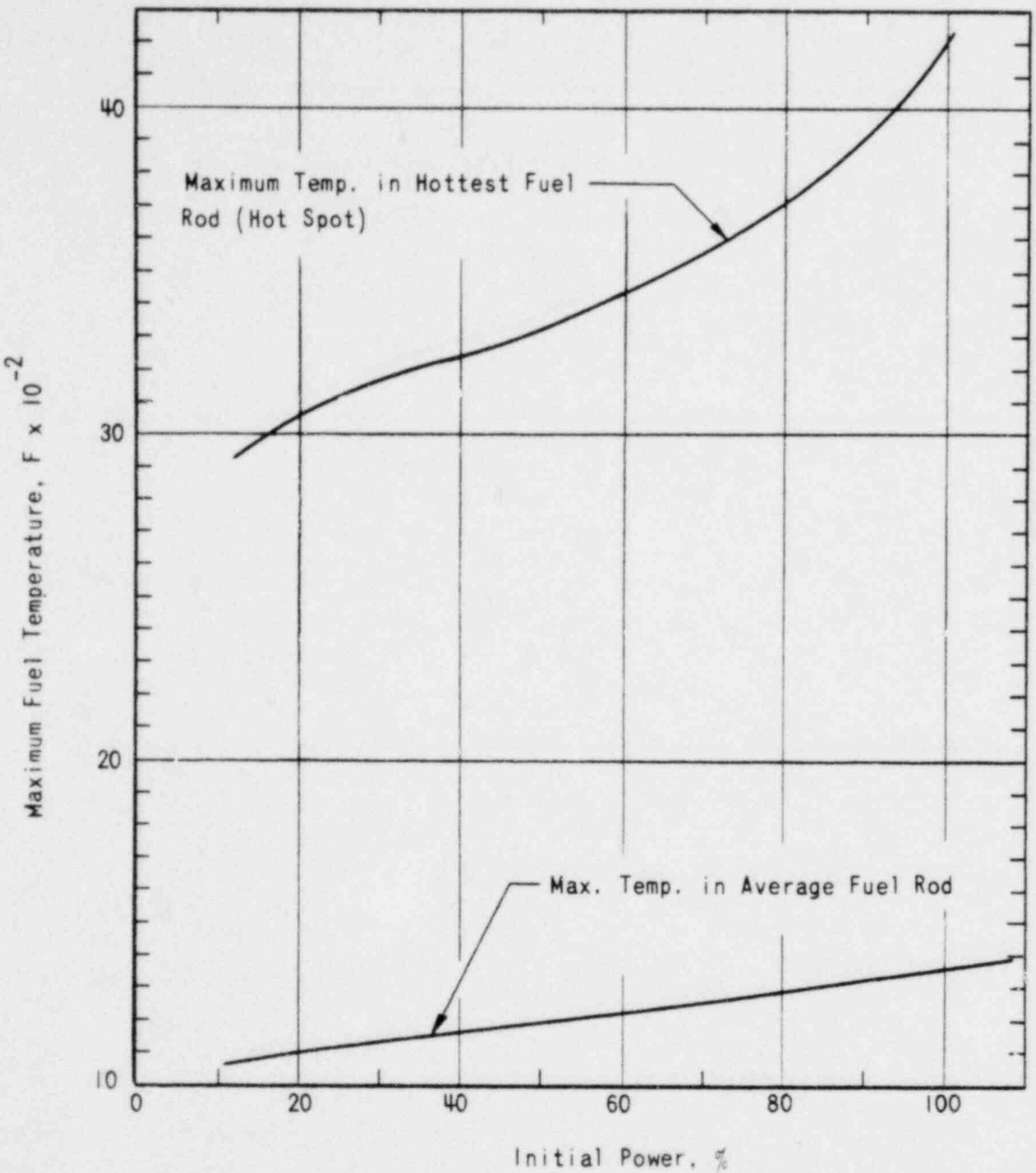
0088

MAXIMUM NEUTRON AND THERMAL POWER FOR AN ALL-ROD WITHDRAWAL ACCIDENT FROM VARIOUS INITIAL POWER LEVELS

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-17



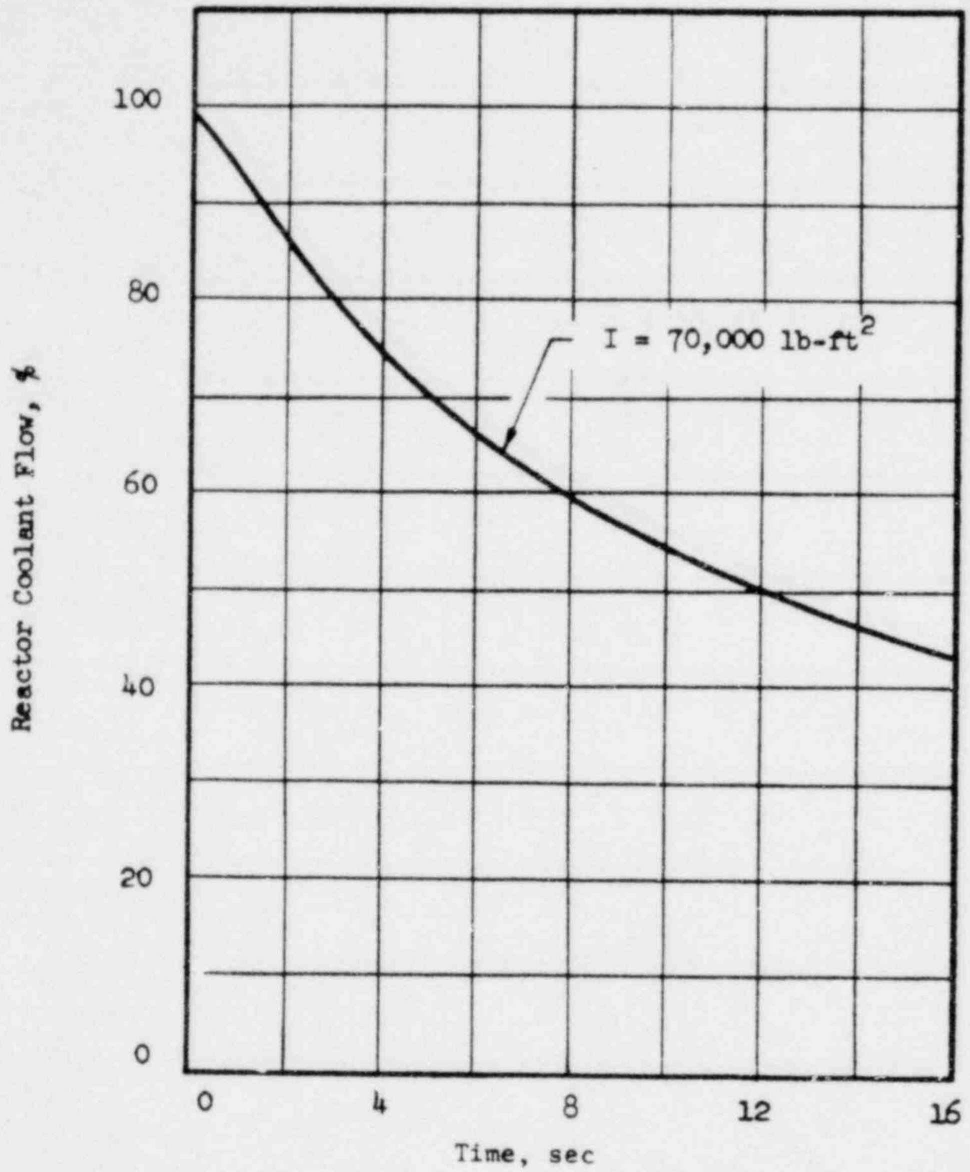
PEAK FUEL TEMPERATURE IN AVERAGE ROD AND HOT SPOT FOR AN ALL-ROD WITHDRAWAL ACCIDENT FROM VARIOUS INITIAL POWER LEVELS

0089

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-18



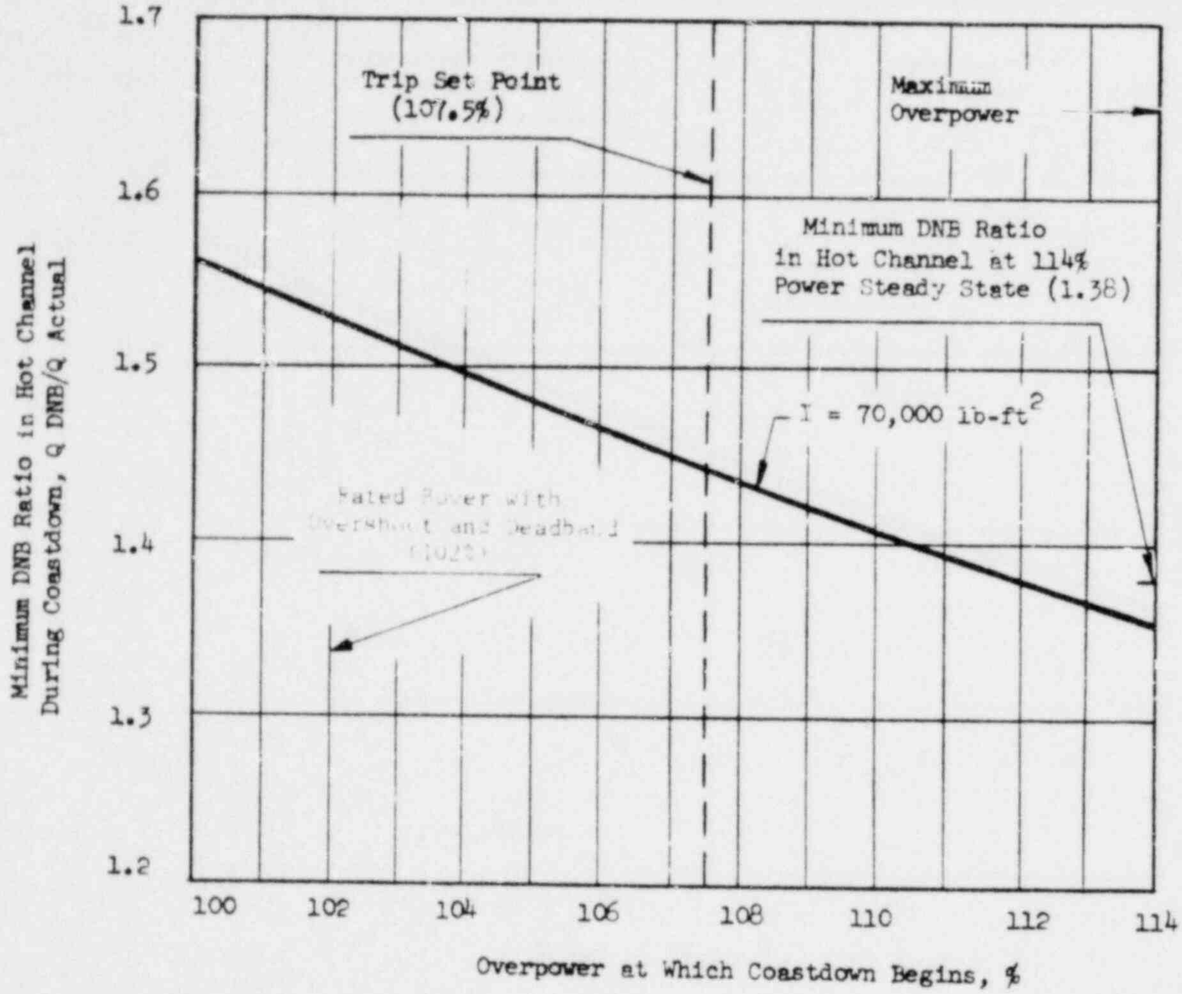
0090

PERCENT REACTOR COOLANT FLOW AS A FUNCTION OF TIME AFTER LOSS OF PUMP POWER

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-19

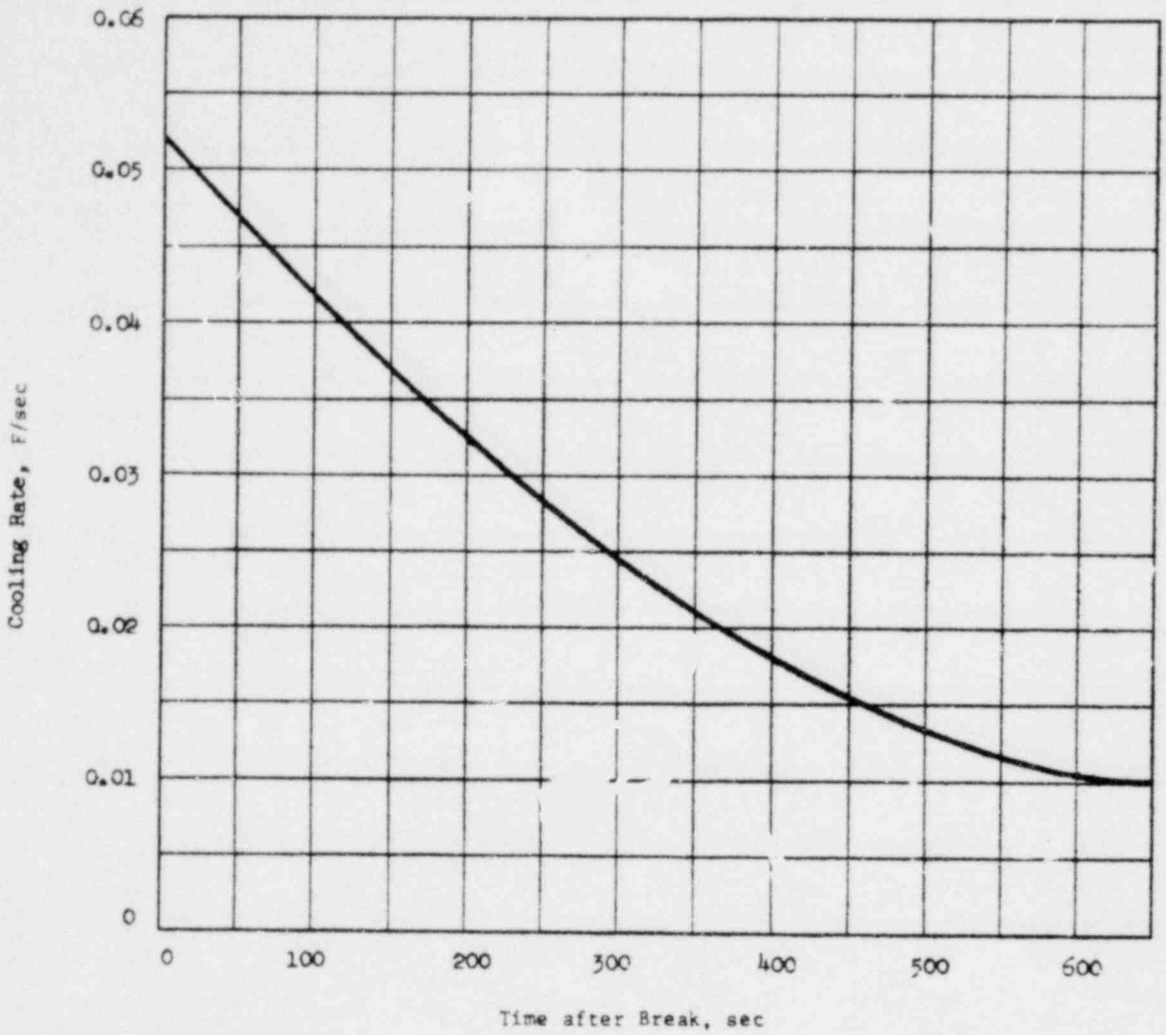


MINIMUM DNBR WHICH OCCURS DURING THE COASTDOWN FOR VARIOUS INITIAL POWER LEVELS

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-20

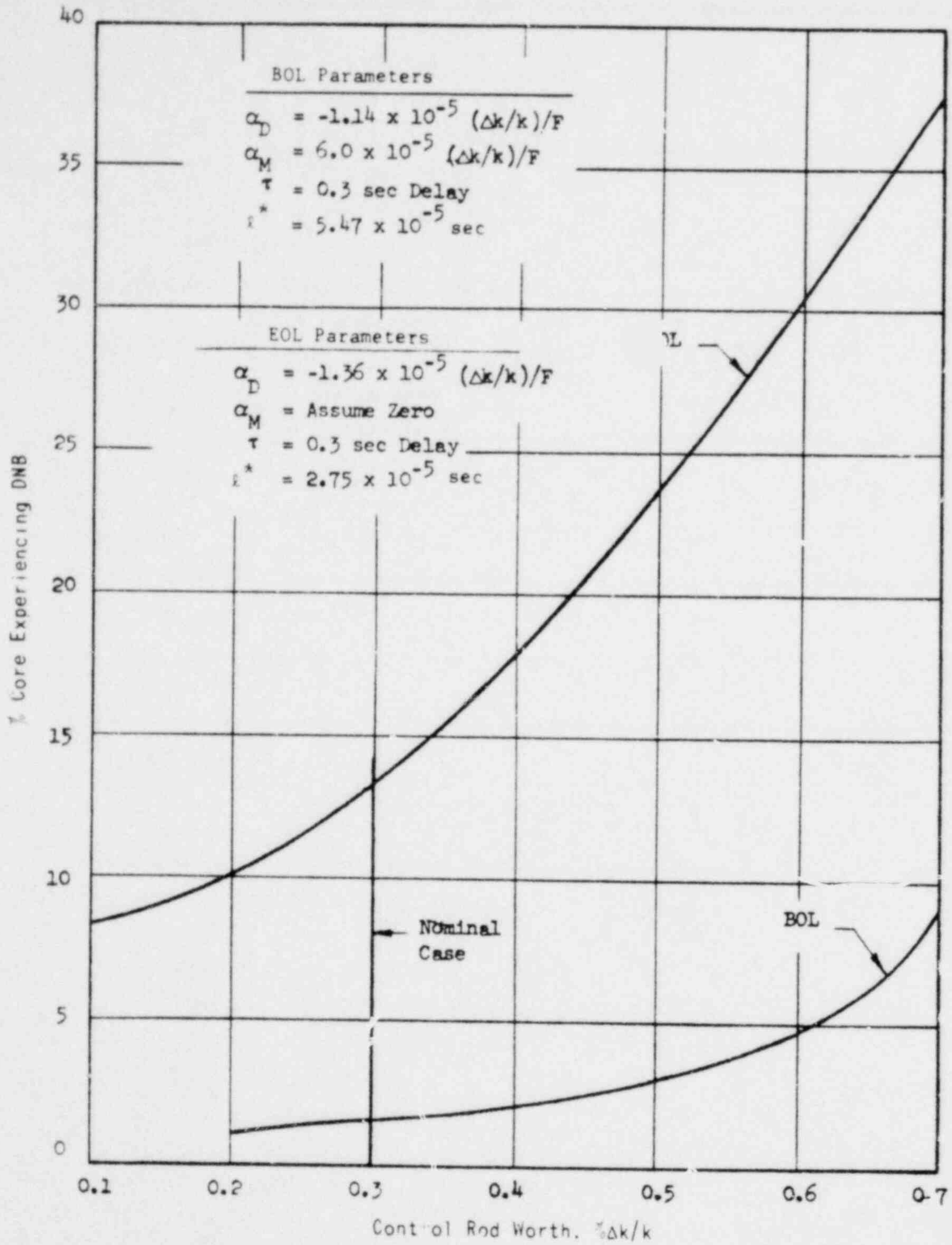


REACTOR SYSTEM COOLING RATE FOR 4 IN.^{2} STEAM LINE BREAK

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-21

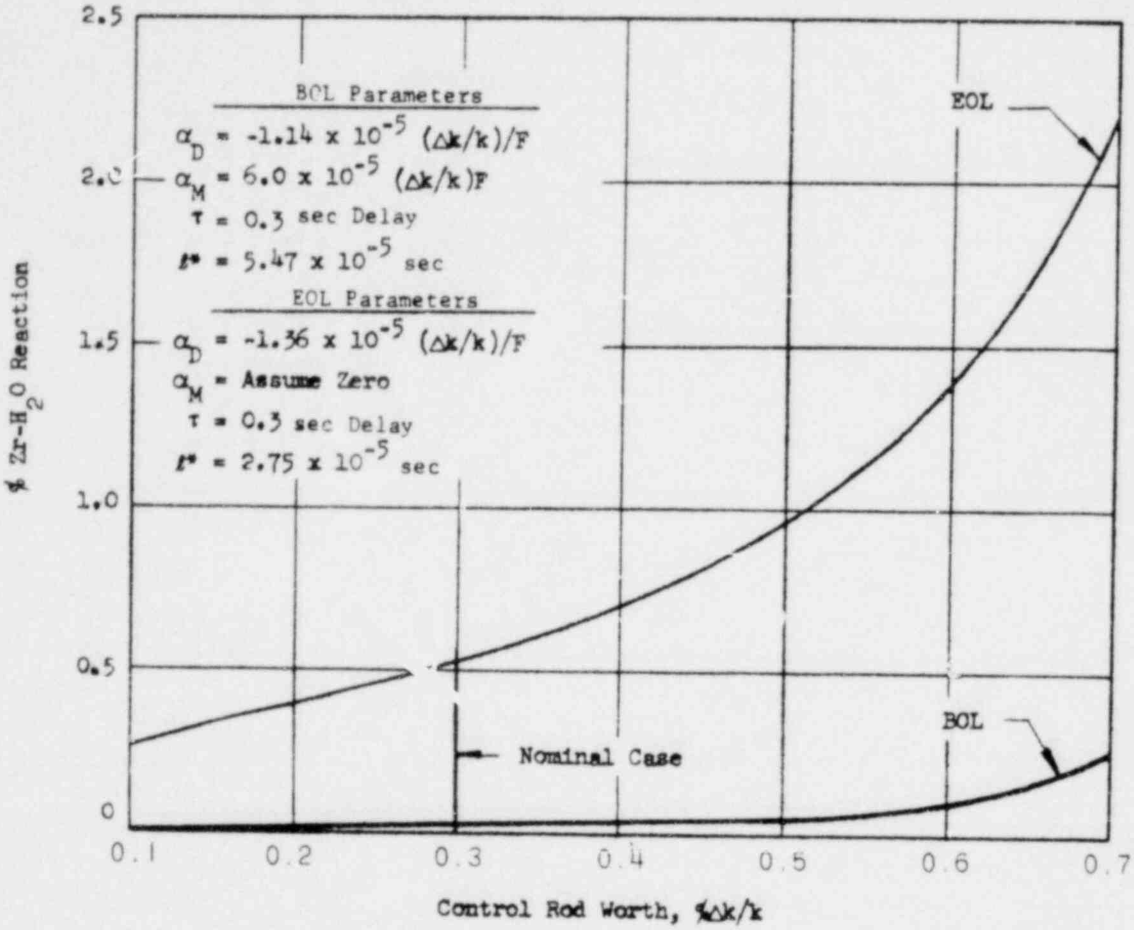


PERCENT CORE EXPERIENCING DNB AS A FUNCTION OF EJECTED CONTROL ROD WORTH AT ULTIMATE POWER

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION FIGURE 14-22

0093



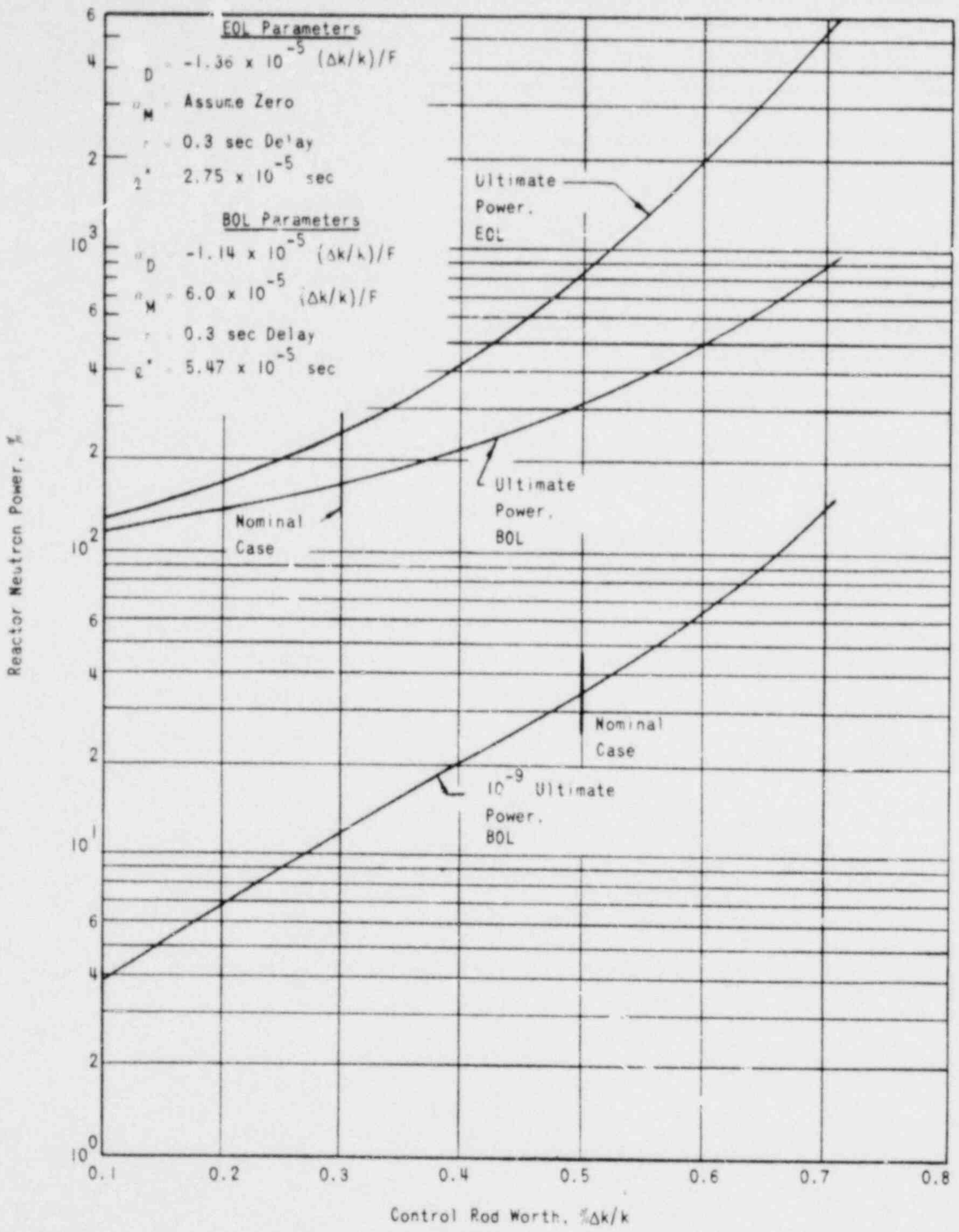
ZR-H_2O REACTION AS A FUNCTION OF EJECTED CONTROL ROD WORTH AT ULTIMATE POWER

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-23

0094

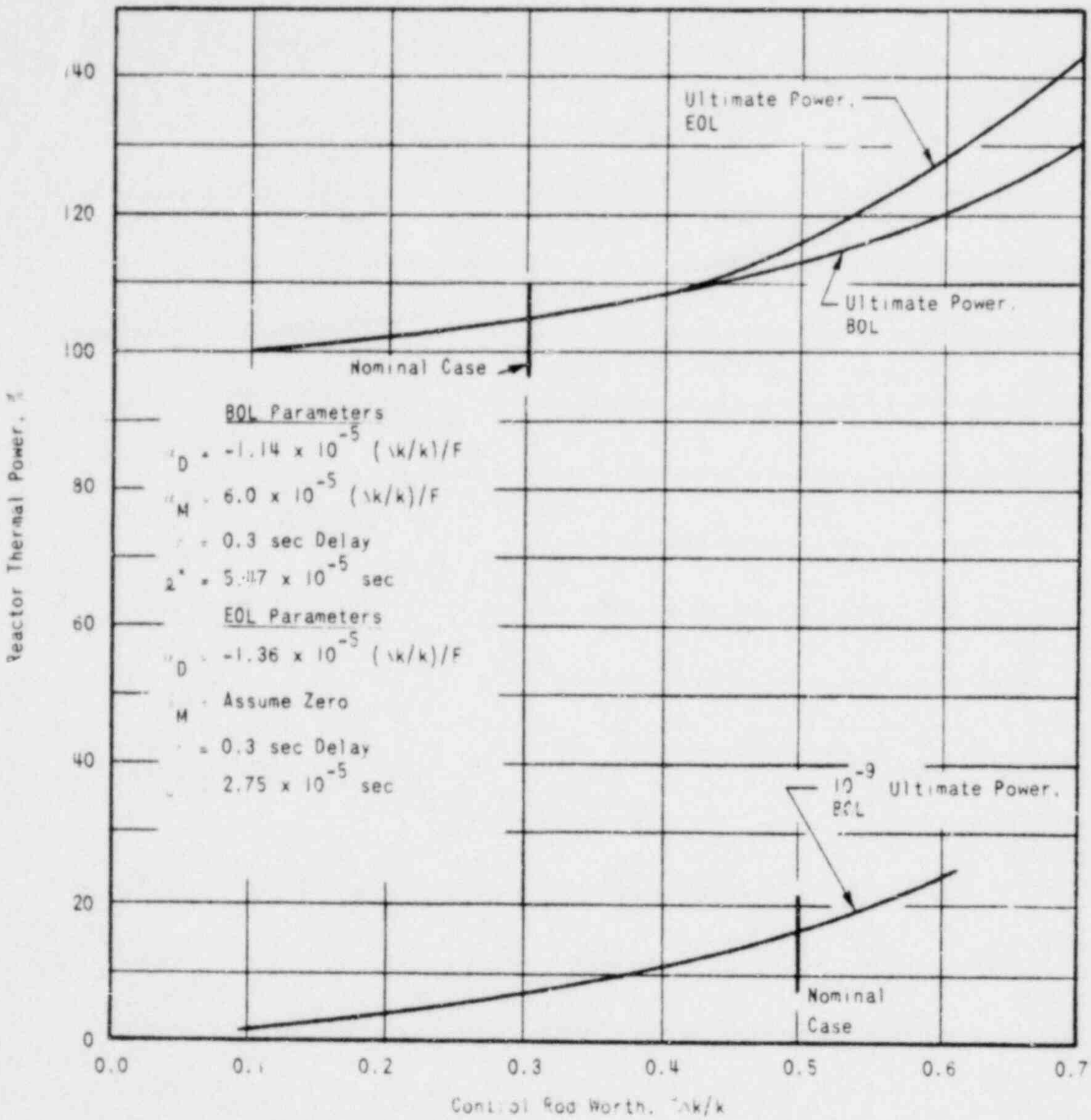


REACTOR NEUTRON POWER VARIATION WITH EJECTED CONTROL ROD WORTH

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-24



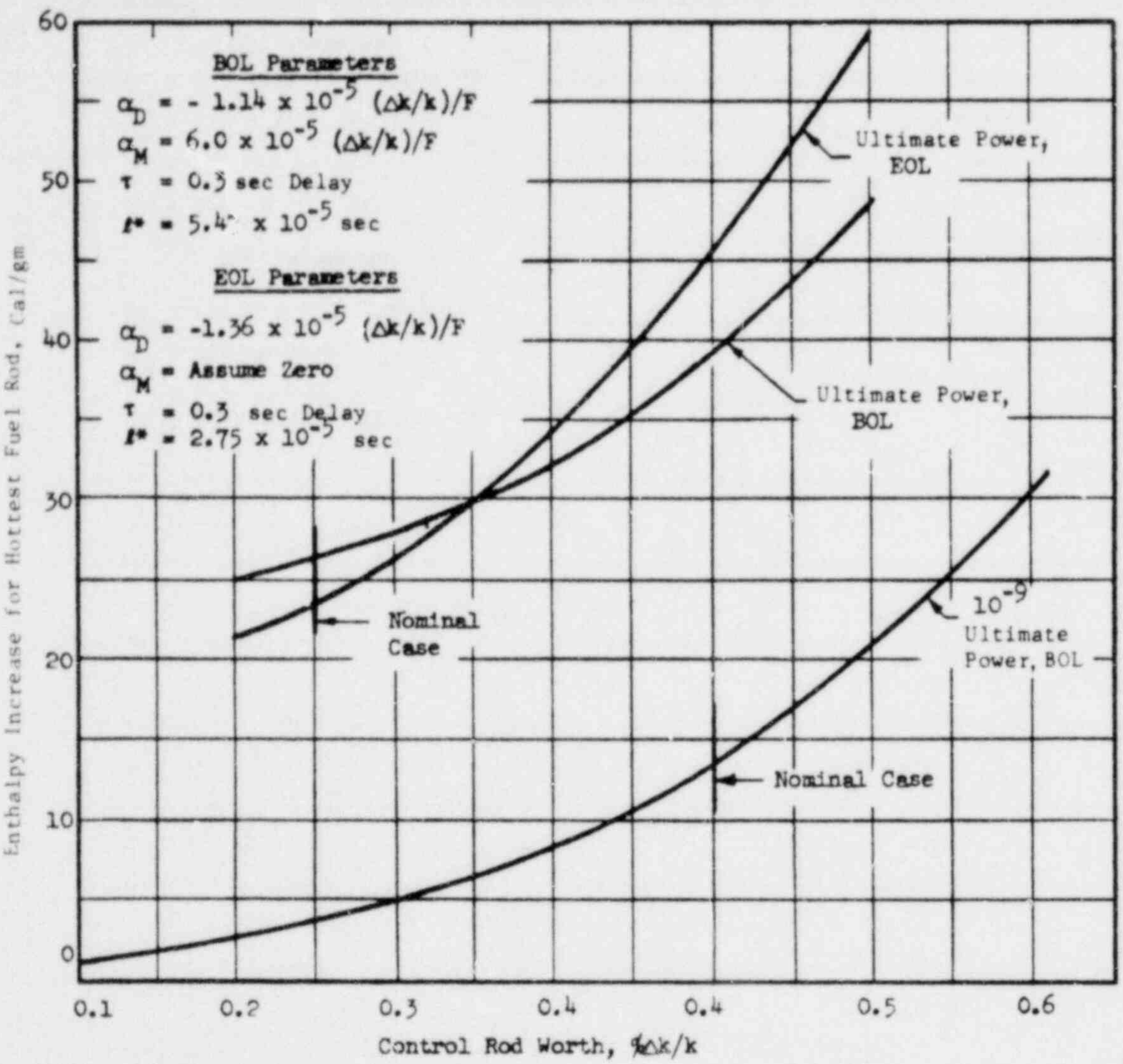
0096

REACTOR THERMAL POWER AS A FUNCTION OF EJECTED CONTROL ROD WORTH

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-25

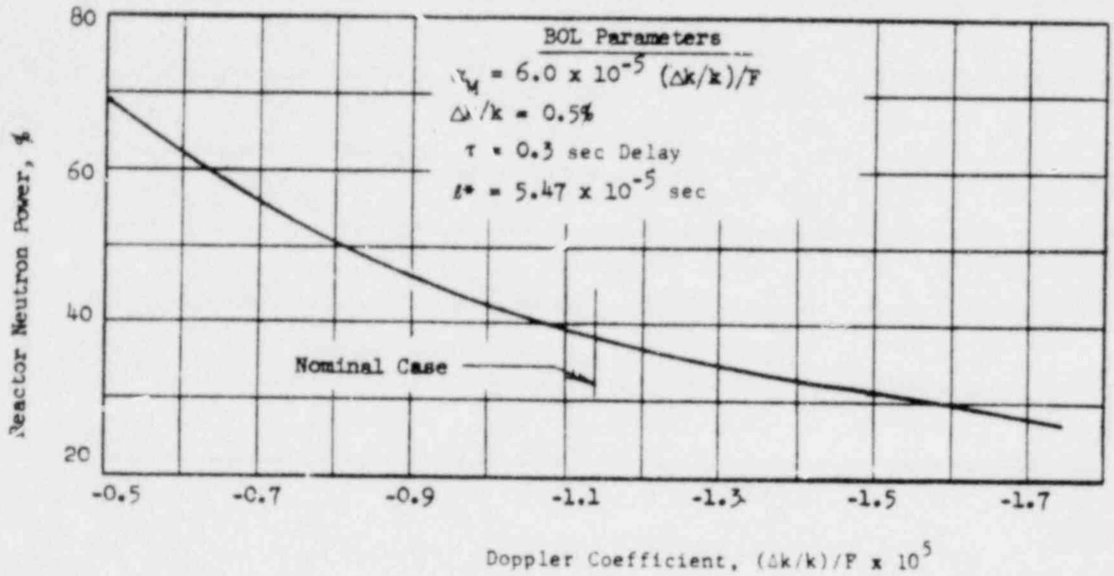


ENTHALPY INCREASE TO THE HOTTEST FUEL ROD VERSUS EJECTED CONTROL ROD WORTH

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-26



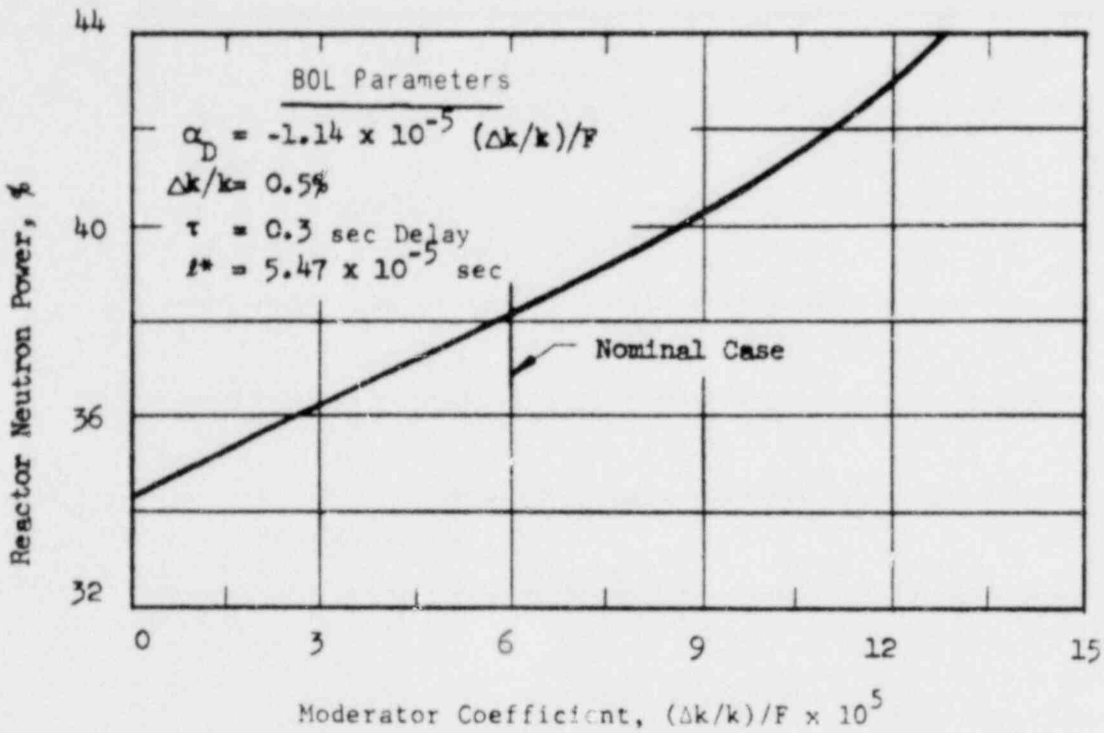
THE EFFECT ON REACTOR NEUTRON POWER OF VARYING THE DOPPLER COEFFICIENT ROD EJECTION AT 10^{-9} ULTIMATE POWER

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-27

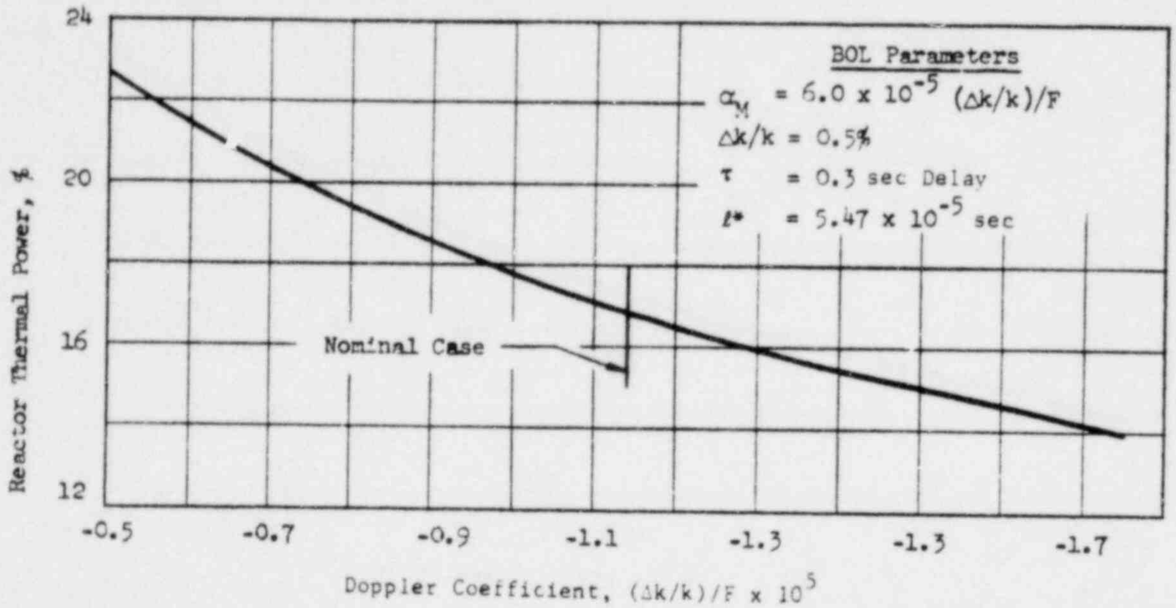
0098



THE EFFECT ON REACTOR NEUTRON POWER OF VARYING THE MODERATOR COEFFICIENT ROD EJECTION AT 10^{-9} ULTIMATE POWER

CRYSTAL RIVER UNITS 3 & 4

FIGURE 14-28

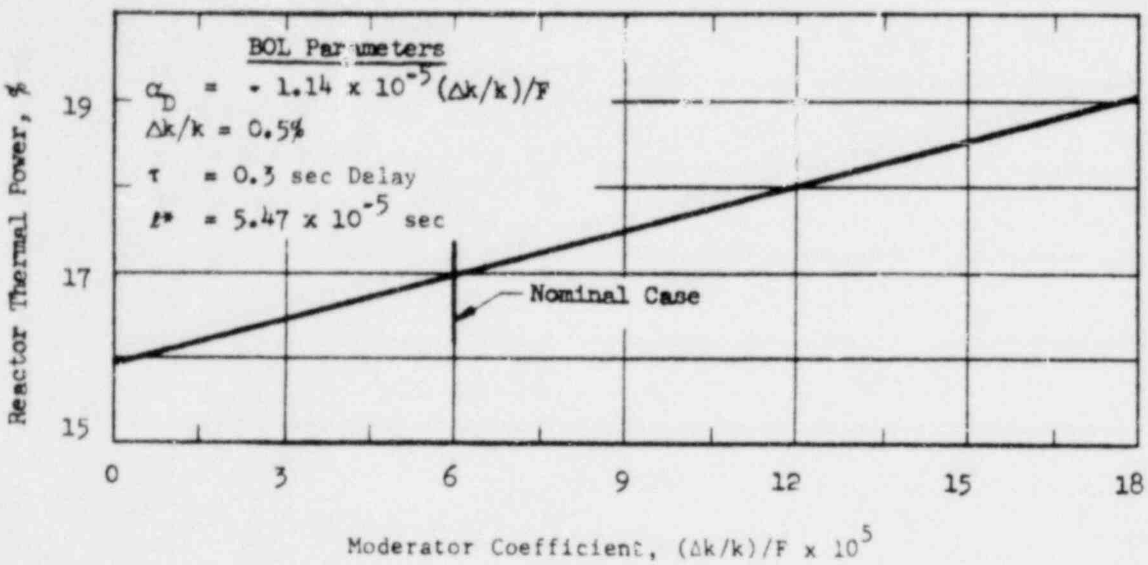


THE EFFECT ON REACTOR THERMAL POWER OF VARYING THE DOPPLER COEFFICIENT ROD EJECTION AT 10^{-9} ULTIMATE POWER

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-29



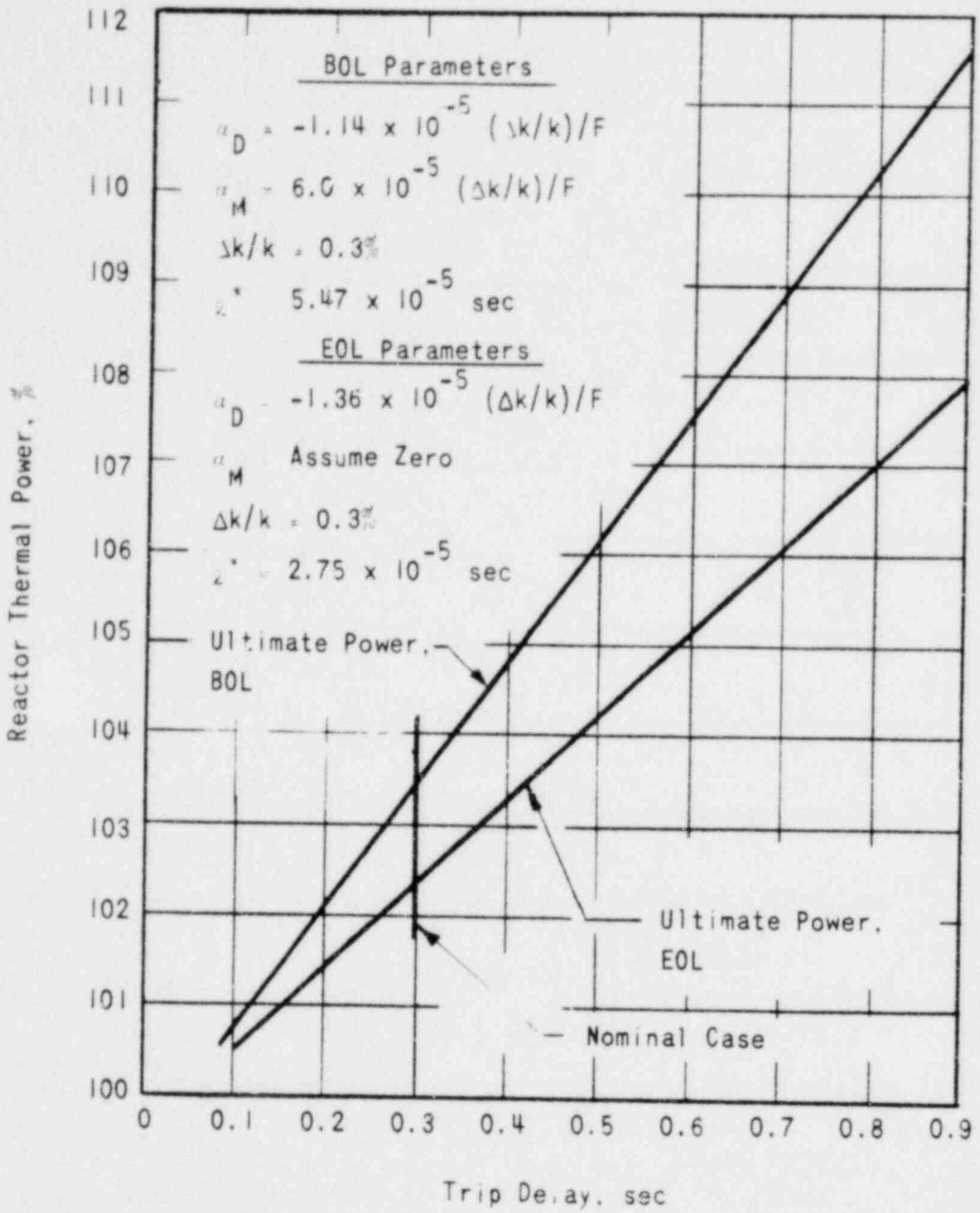
THE EFFECT ON REACTOR THERMAL POWER OF VARYING THE MODERATOR COEFFICIENT ROD EJECTION AT 10^{-9} ULTIMATE POWER

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-30

0101



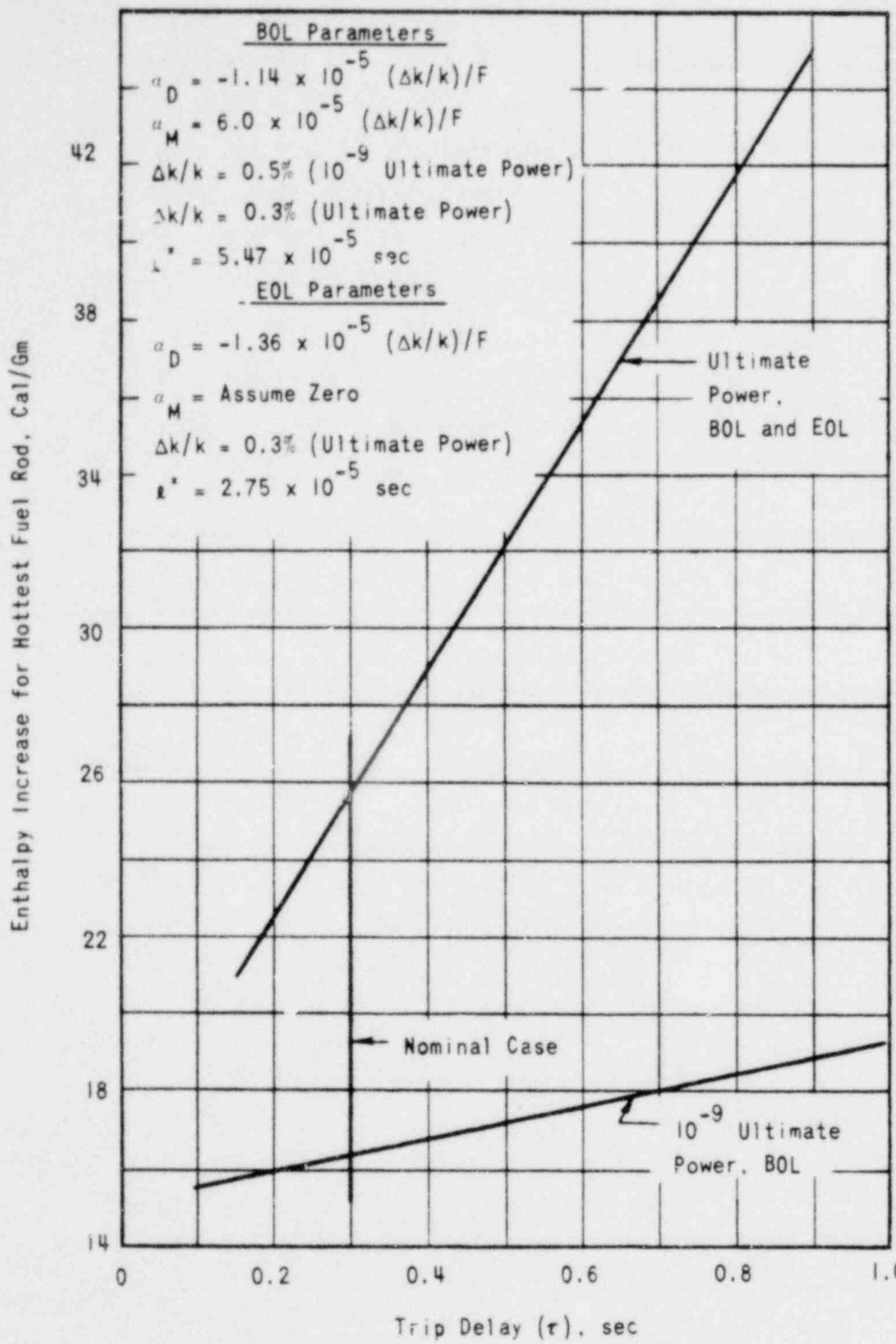
REACTOR THERMAL POWER VERSUS TRIP DELAY TIME - ROD EJECTION AT ULTIMATE POWER

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-31

0102

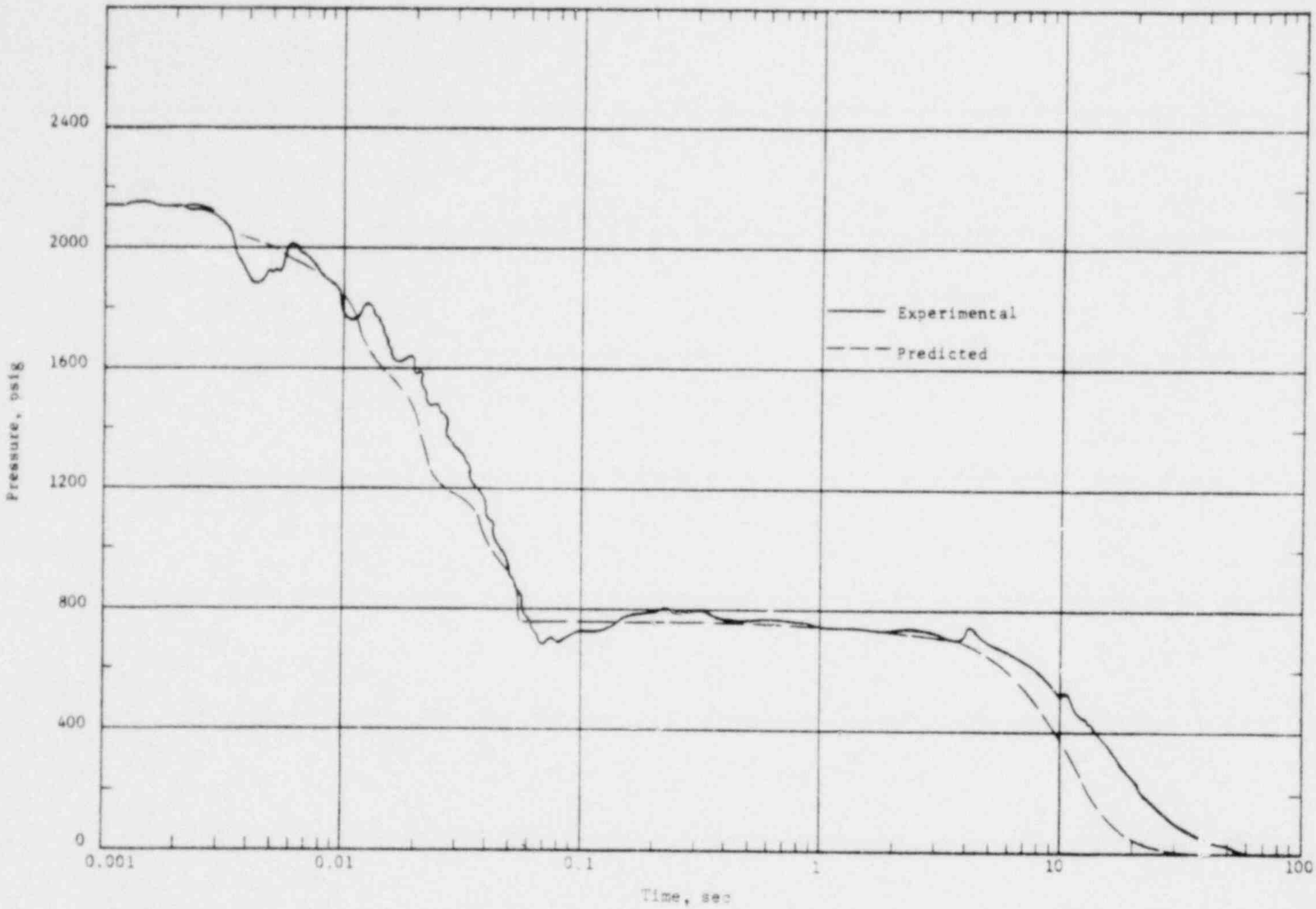


ENTHALPY INCREASE TO THE HOTTEST FUEL ROD VERSUS TRIP DELAY TIME - ROD EJECTION

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-32



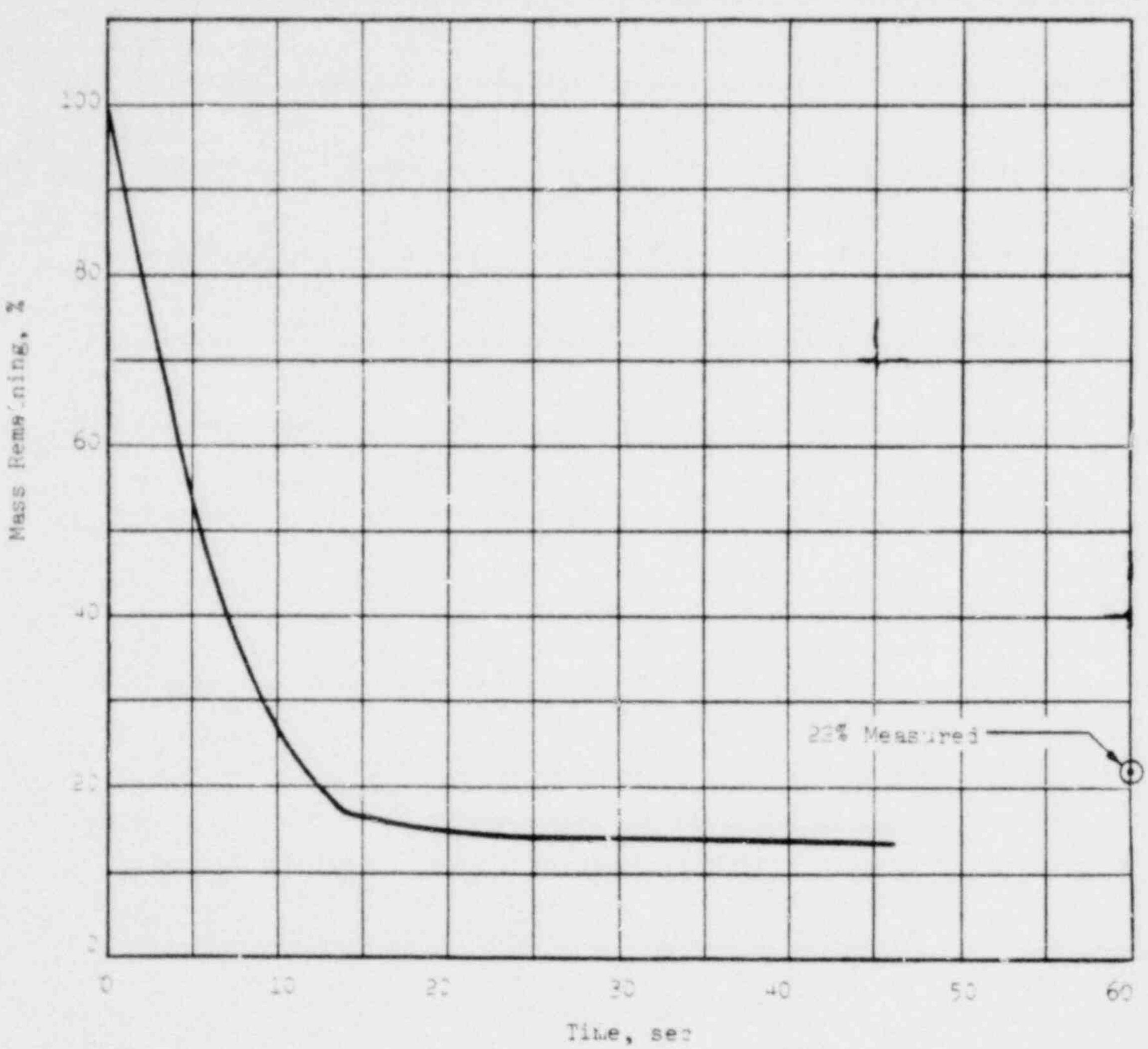
LOFT SEMISCALE BLOWDOWN TEST NO. 546
VESSEL PRESSURE VERSUS TIME

0104

CRYSTAL RIVER UNITS 3 & 4

FPC FLORIDA POWER CORPORATION

FIGURE 14-33

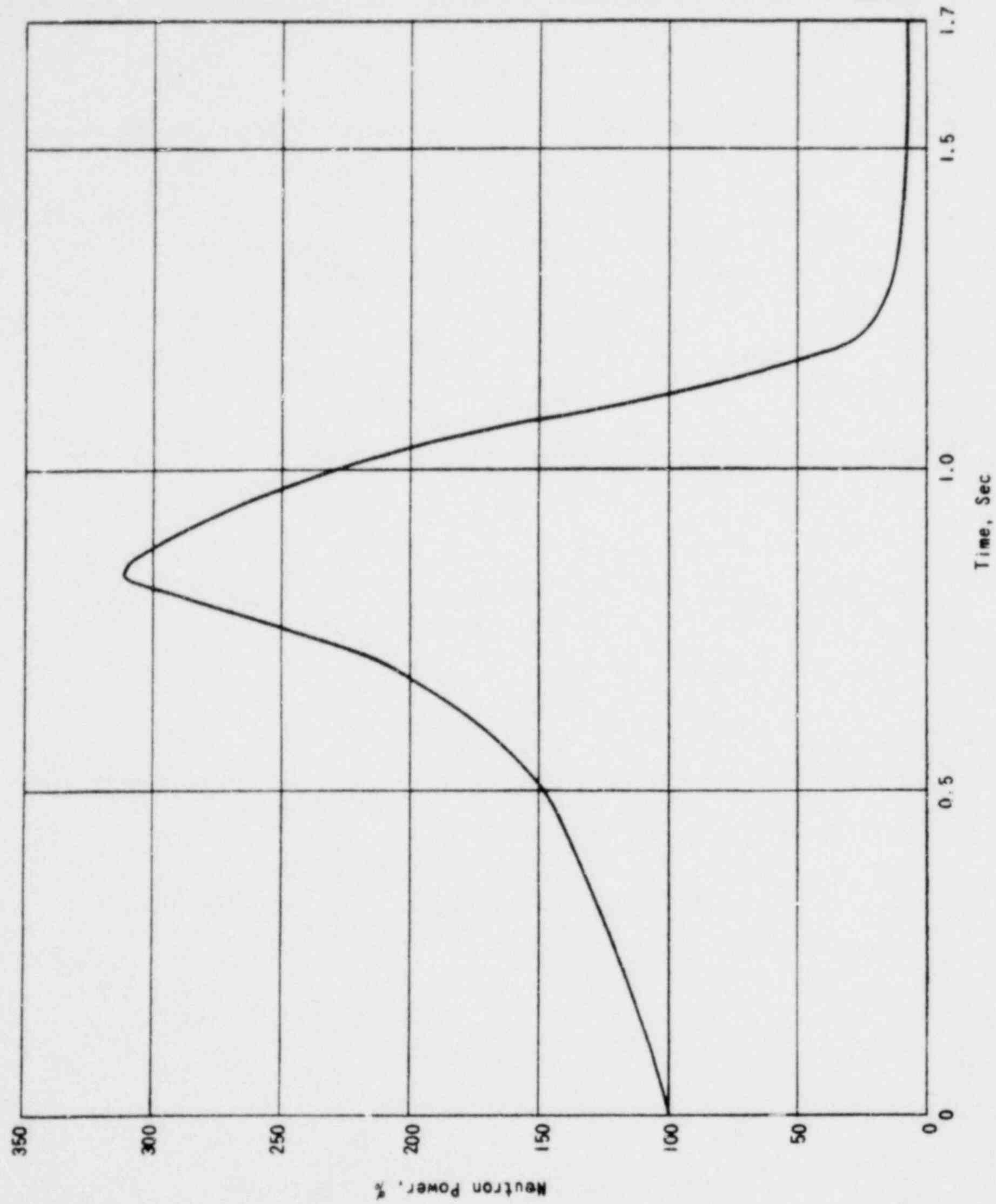


0105

PREDICTED PERCENT MASS REMAINING VERSUS TIME
LOFT TEST NO. 546

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION FIGURE 14-34



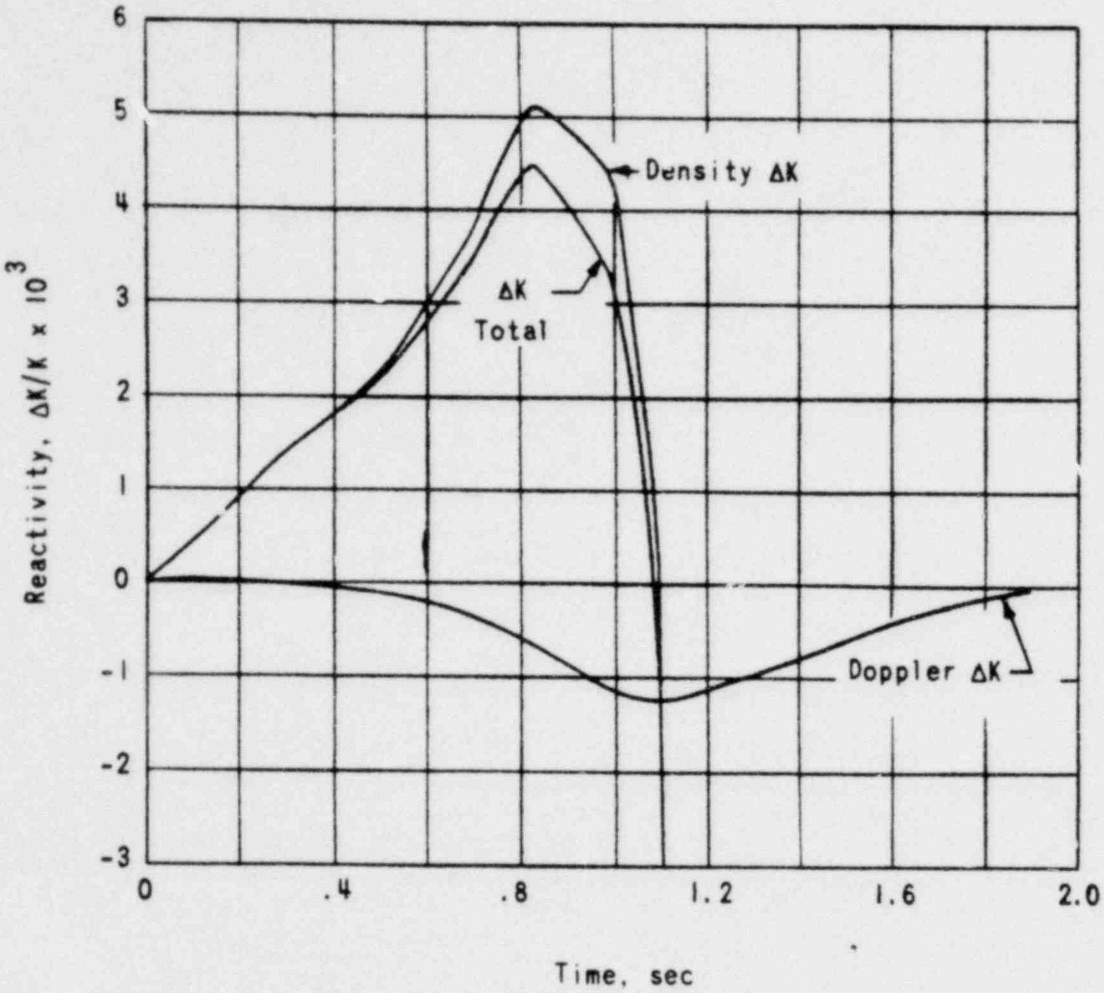
NEUTRON POWER VERSUS TIME FOR A 36 IN. ID, DOUBLE-ENDED, HOT LEG PIPE RUPTURE AT ULTIMATE POWER WITHOUT TRIP

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION FIGURE 14 - 34 - a

AMEND. 1 (1-15-68)

0106



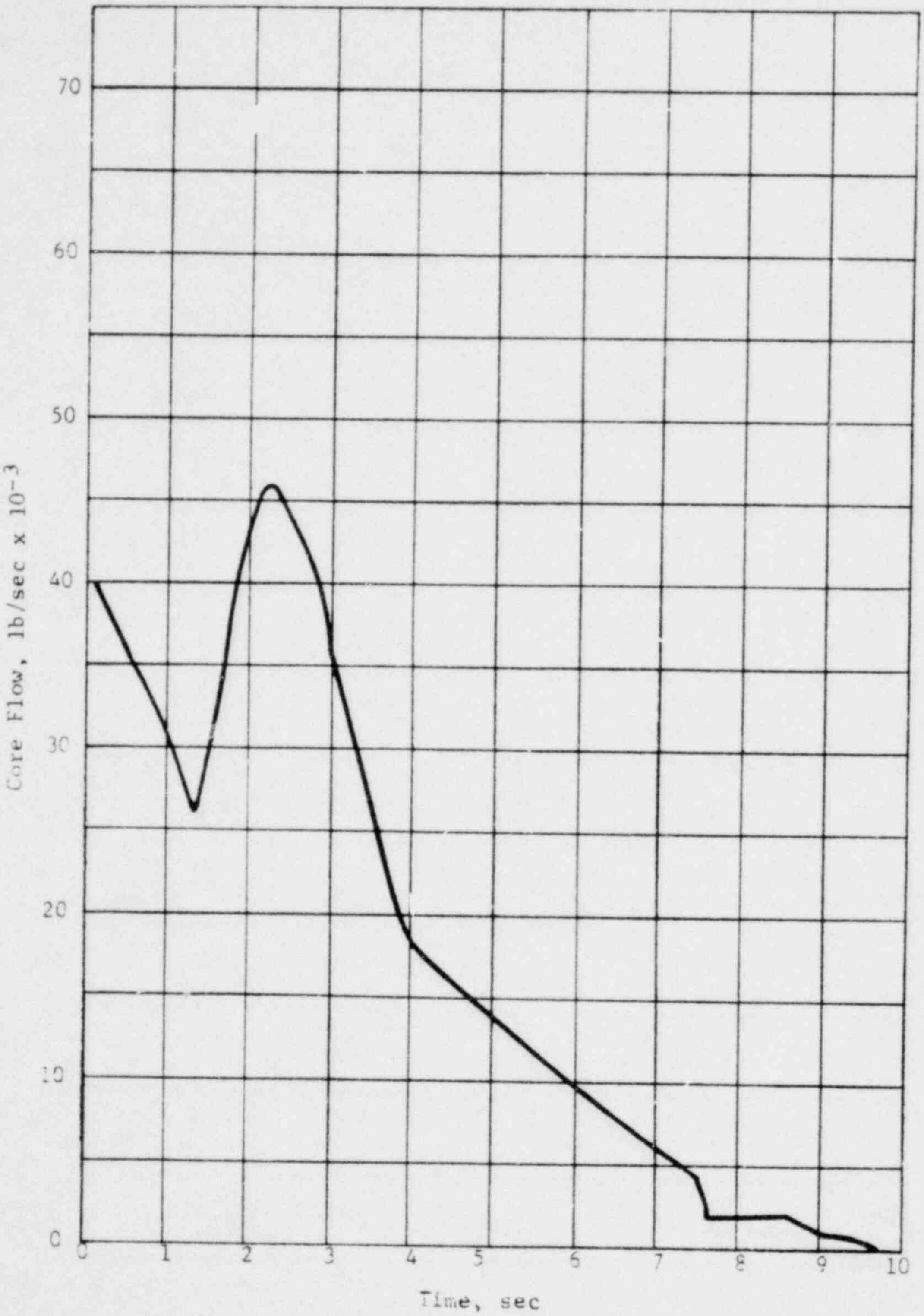
REACTIVITY VERSUS TIME FOR A 36 IN.
ID, DOUBLE-ENDED, HOT LEG PIPE RUPTURE
AT ULTIMATE POWER WITHOUT TRIP

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION FIGURE 14-34-b

AMEND. 1 (1-15-68)

0107

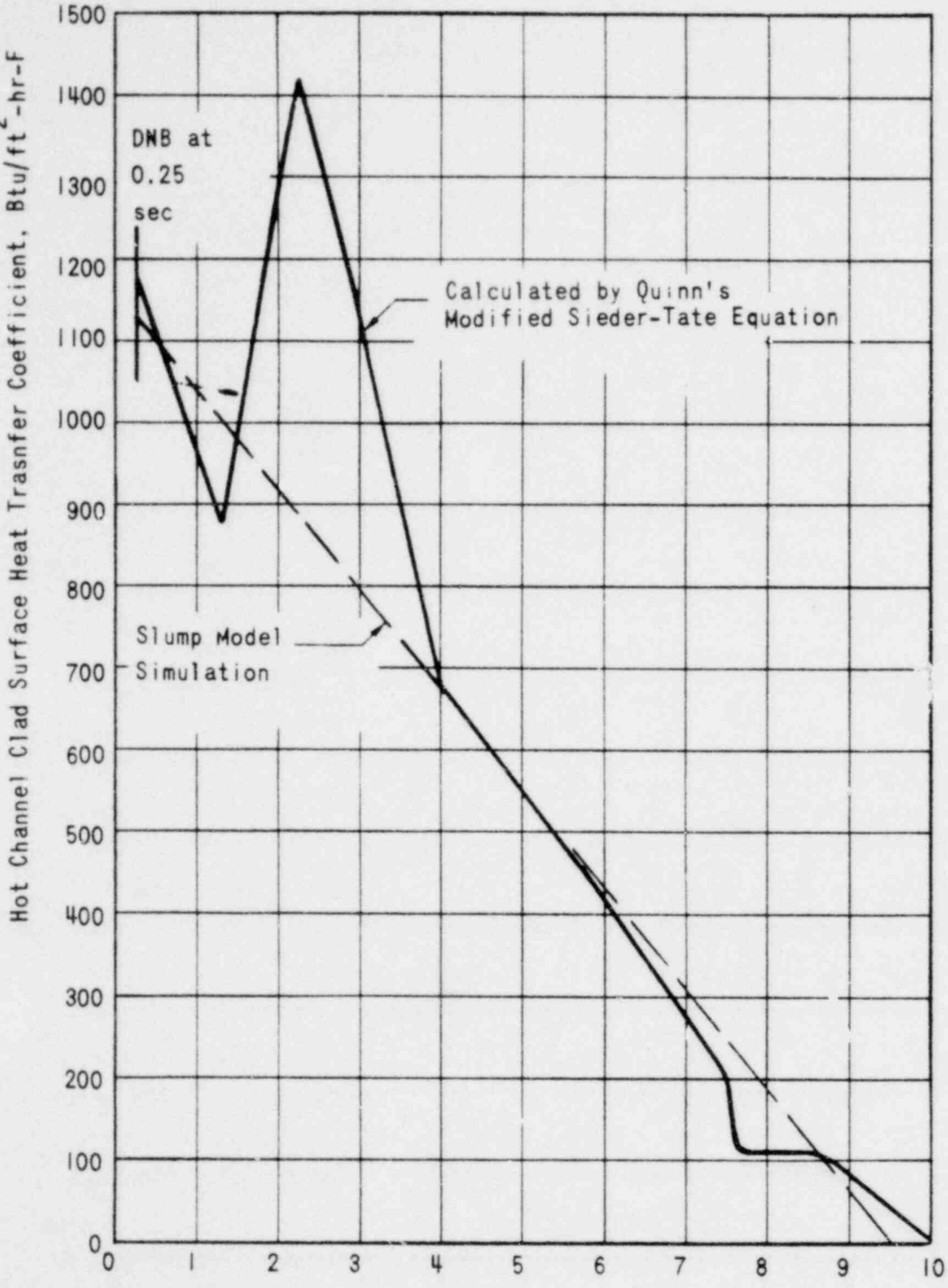


CORE FLOW VERSUS TIME FOR A 36 IN. ID, DOUBLE-ENDED PIPE RUPTURE

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-35



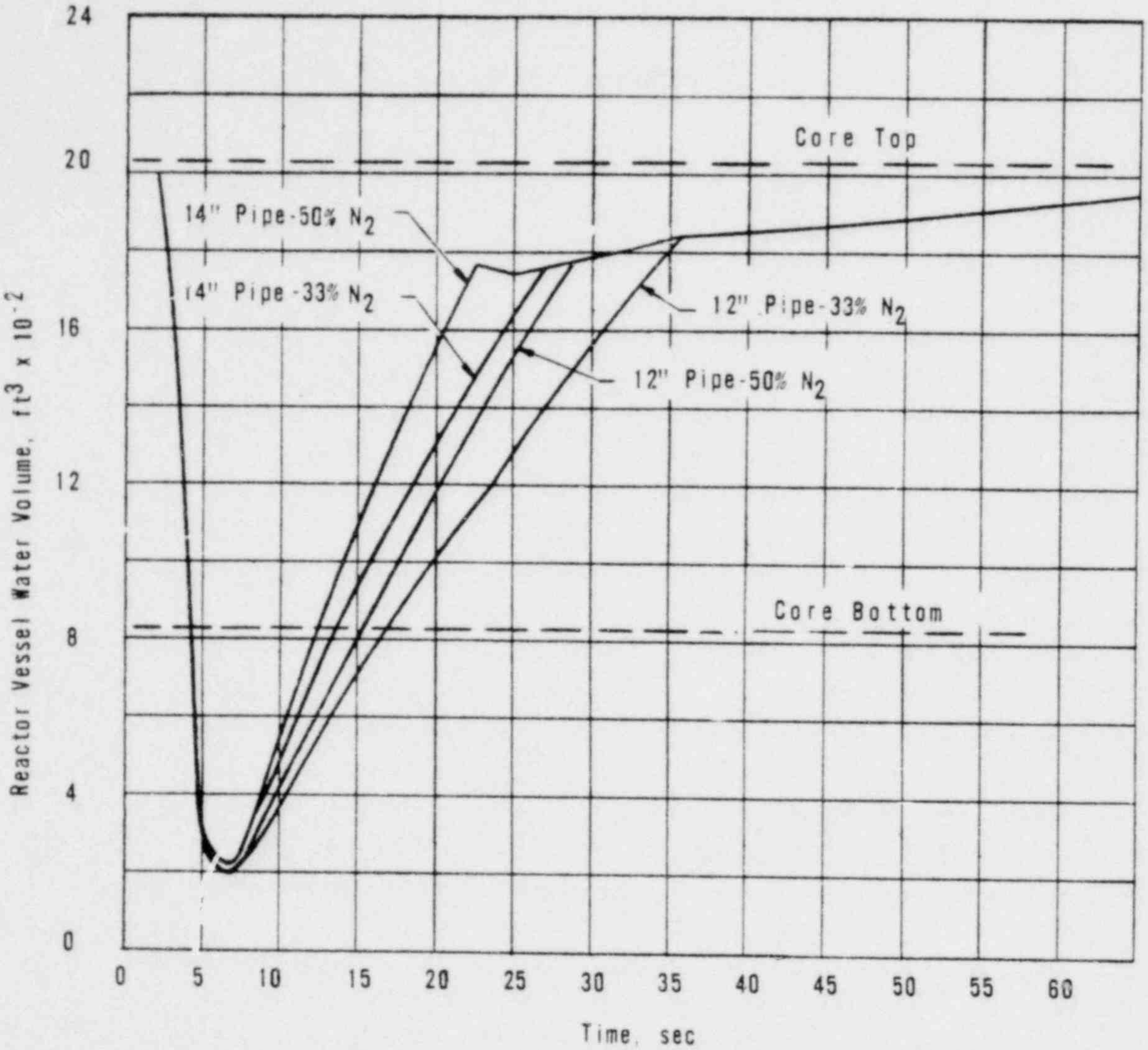
HOT CHANNEL CLAD SURFACE HEAT TRANSFER
COEFFICIENT AFTER DNB VERSUS TIME
FOR A 36 IN. ID, DOUBLE-ENDED PIPE RUPTURE

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-36

0109



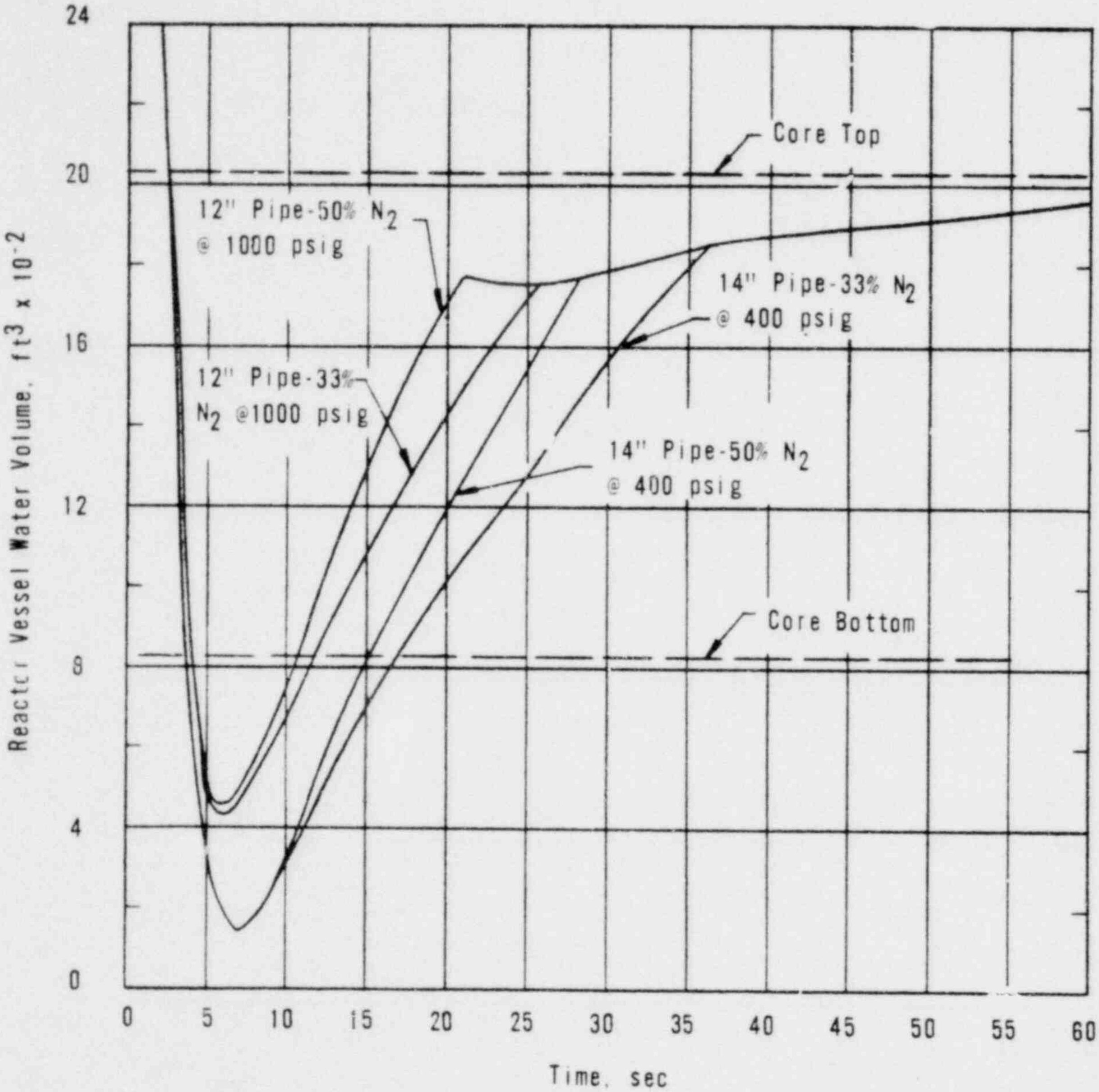
REACTOR VESSEL WATER VOLUME VERSUS TIME FOR 36 IN. ID, DOUBLE-ENDED PIPE RUPTURE FOR 600 PSIG CORE FLOODING TANK OPERATING PRESSURE

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION FIGURE 14-37

AMEND. 2 (2-7-68)

0110



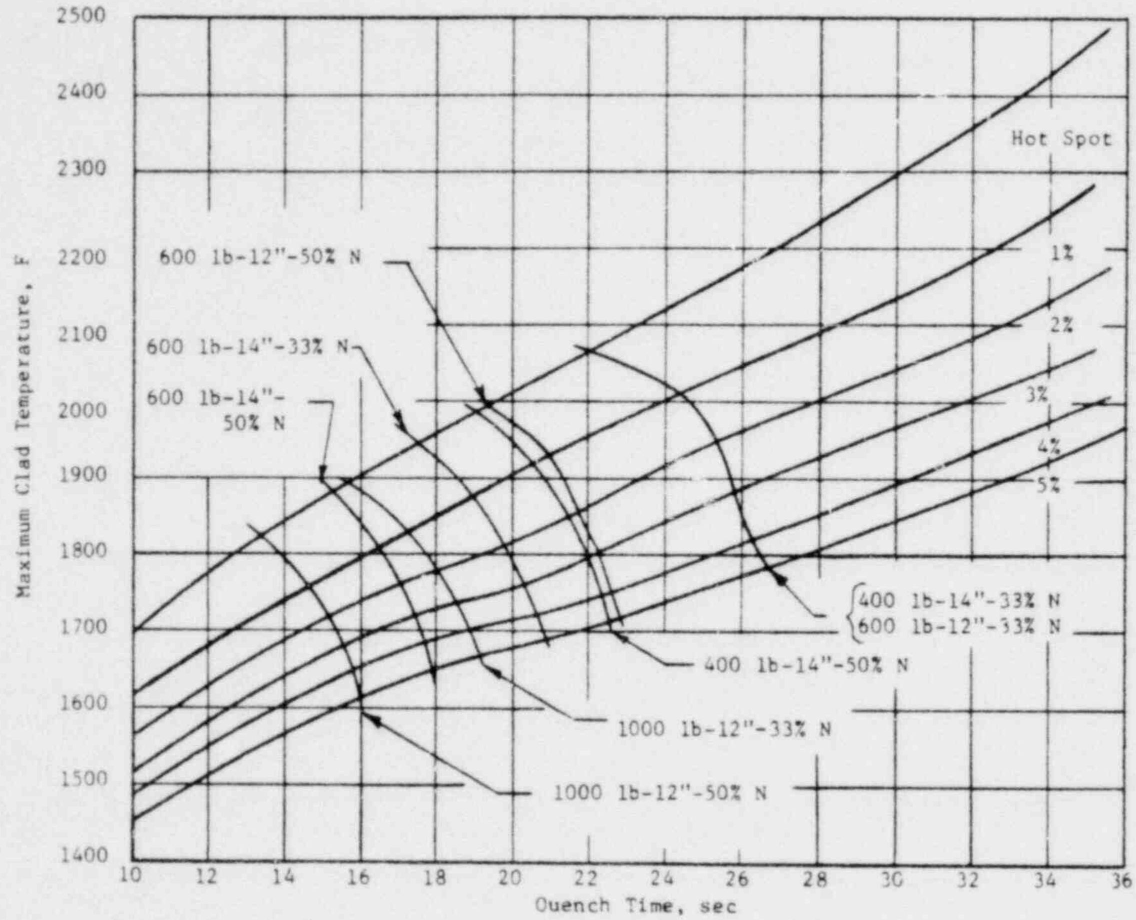
REACTOR VESSEL WATER VOLUME VERSUS TIME FOR
36 IN. ID, DOUBLE-ENDED PIPE RUPTURE FOR 400 PSIG
AND 1,000 PSIG CORE FLOODING TANK OPERATING PRESSURES

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION FIGURE 14-38

AMEND. 2 (2-7-68)

0111



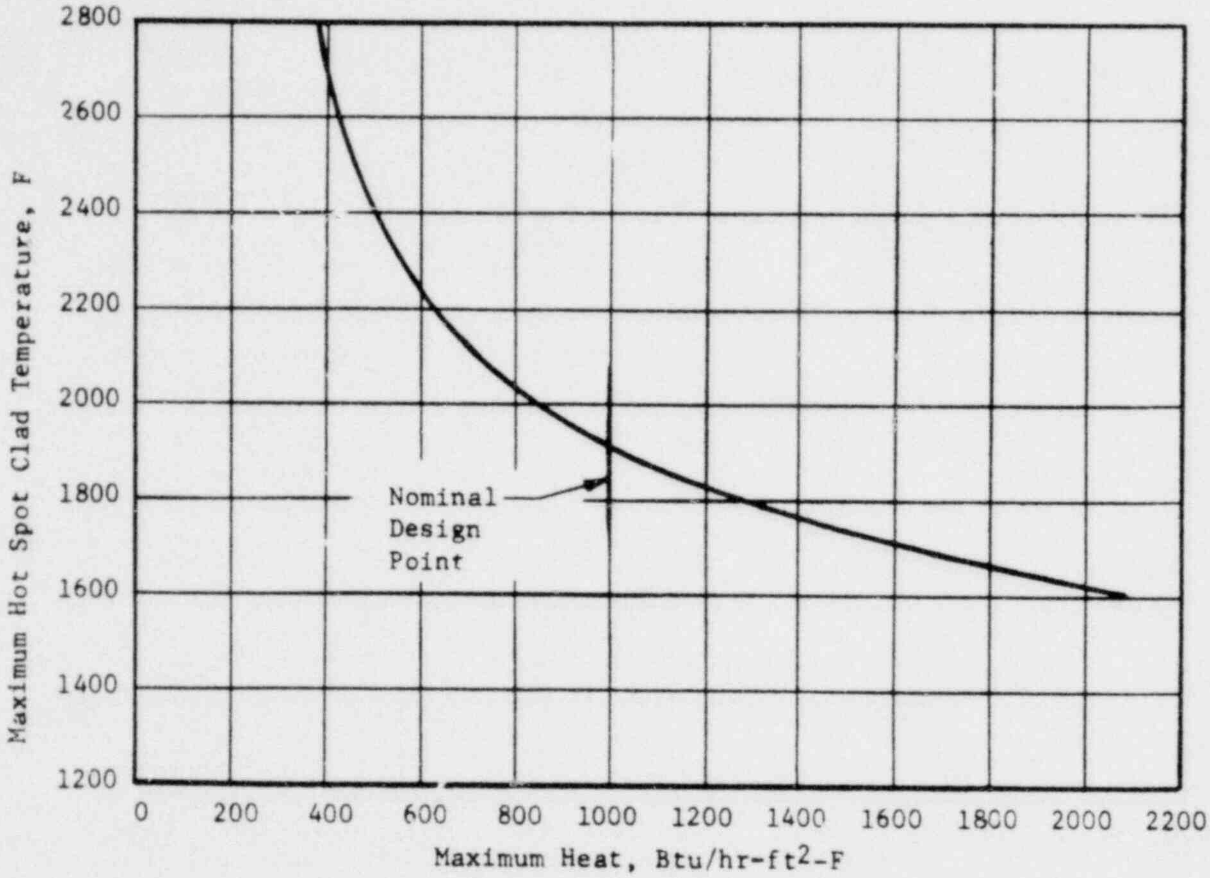
CORE FLOODING TANK ANALYSIS; MAXIMUM CLAD TEMPERATURE VERSUS TIME TO QUENCH FOR A 36 IN. ID, DOUBLE-ENDED PIPE RUPTURE

0112

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-39



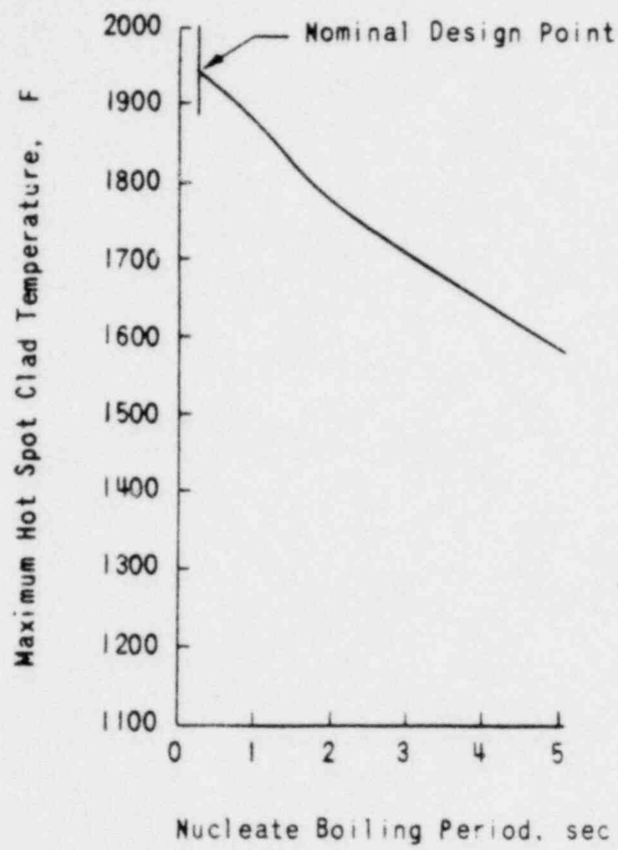
MAXIMUM HOT SPOT CLAD TEMPERATURE VERSUS MAXIMUM HEAT TRANSFER COEFFICIENT AFTE' DNB FOR A 36 IN. ID, DOUBLE-ENDED PIPE RUPT RE

0113

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-40



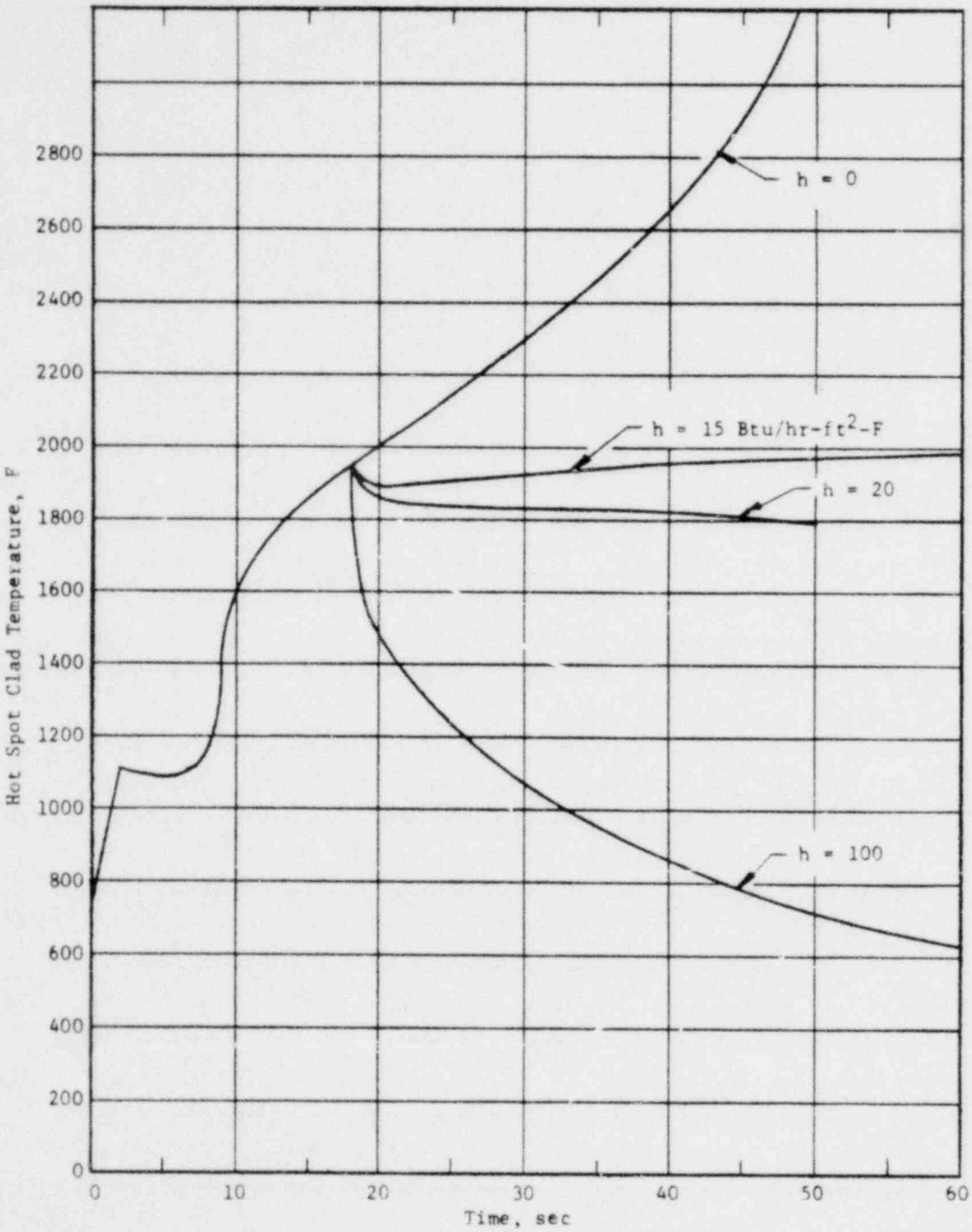
0114

MAXIMUM HOT SPOT CLAD TEMPERATURE AS A FUNCTION OF TIME TO REACH DNB FOR A 36 IN. ID, DOUBLE-ENDED, HOT LEG PIPE RUPTURE

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION FIGURE 14-40

AMEND. 1 (1-15-68)



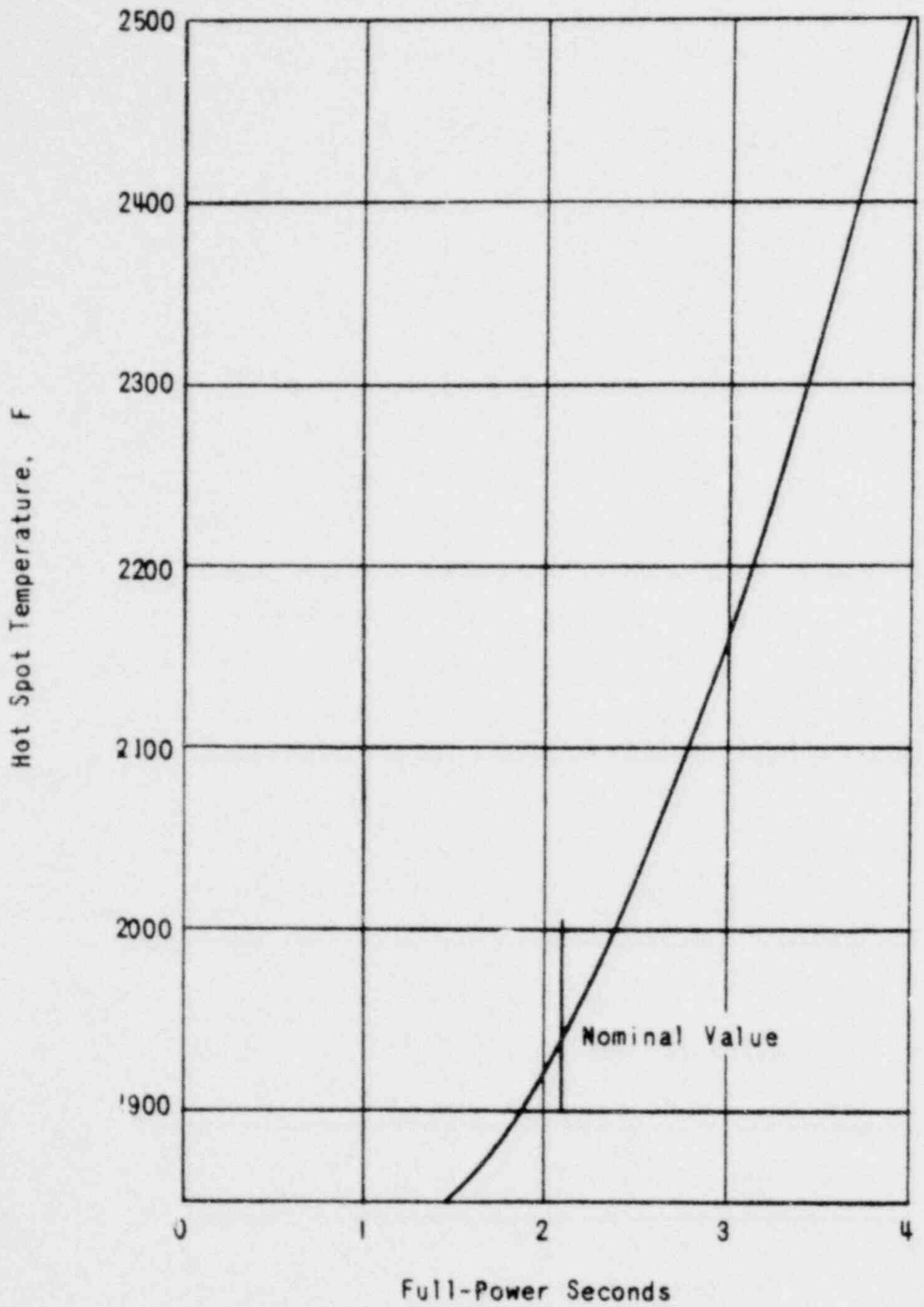
HOT SPOT CLAD TEMPERATURE VERSUS TIME FOR 36 IN. ID, DOUBLE-ENDED PIPE RUPTURE AND VARIABLE QUENCH COEFFICIENT

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-41

0115



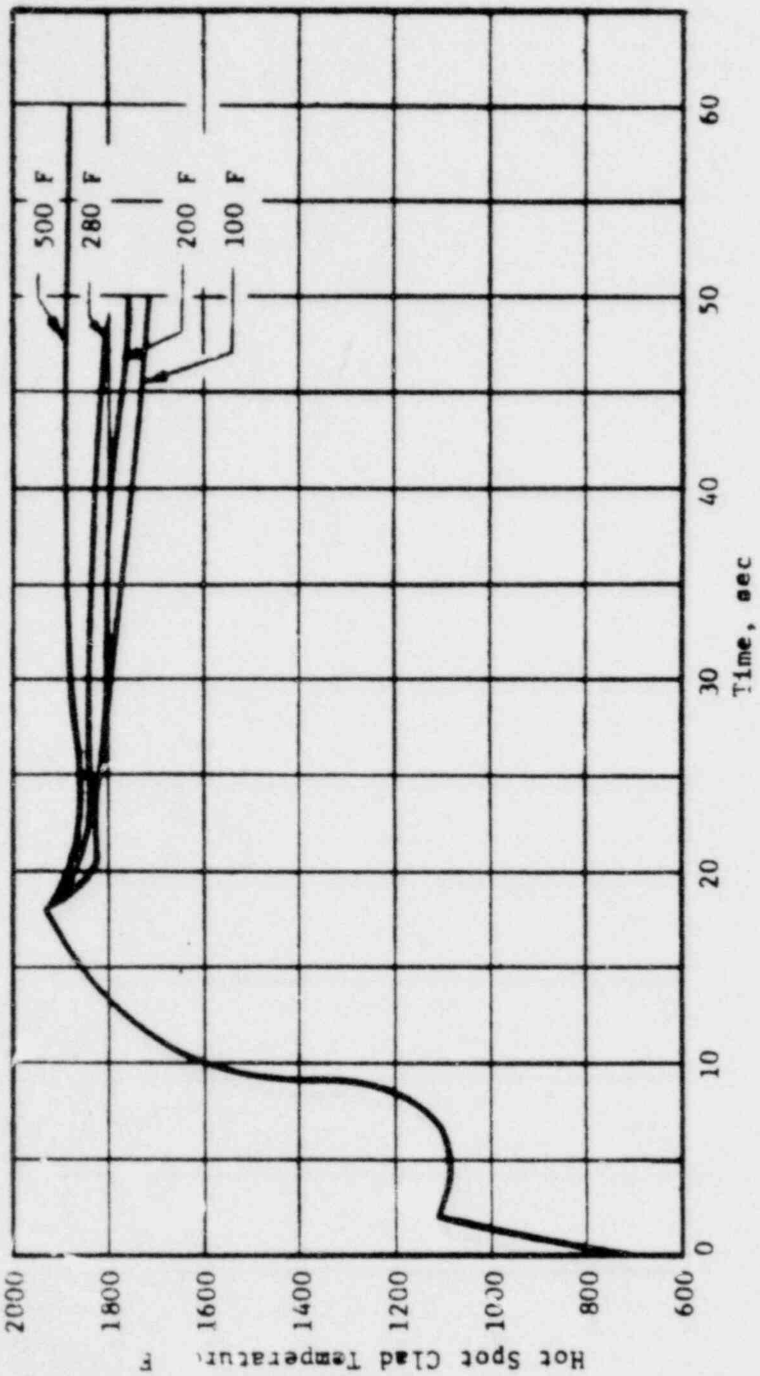
HOT SPOT CLAD TEMPERATURE AS A FUNCTION OF FULL-POWER SECONDS RESULTING FROM VOID SHUTDOWN FOR A 36 IN. ID, DOUBLE-ENDED, HOT LEG PIPE RUPTURE

0116

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION FIGURE 14-41-a

AMEND. 1 (1-15-68)

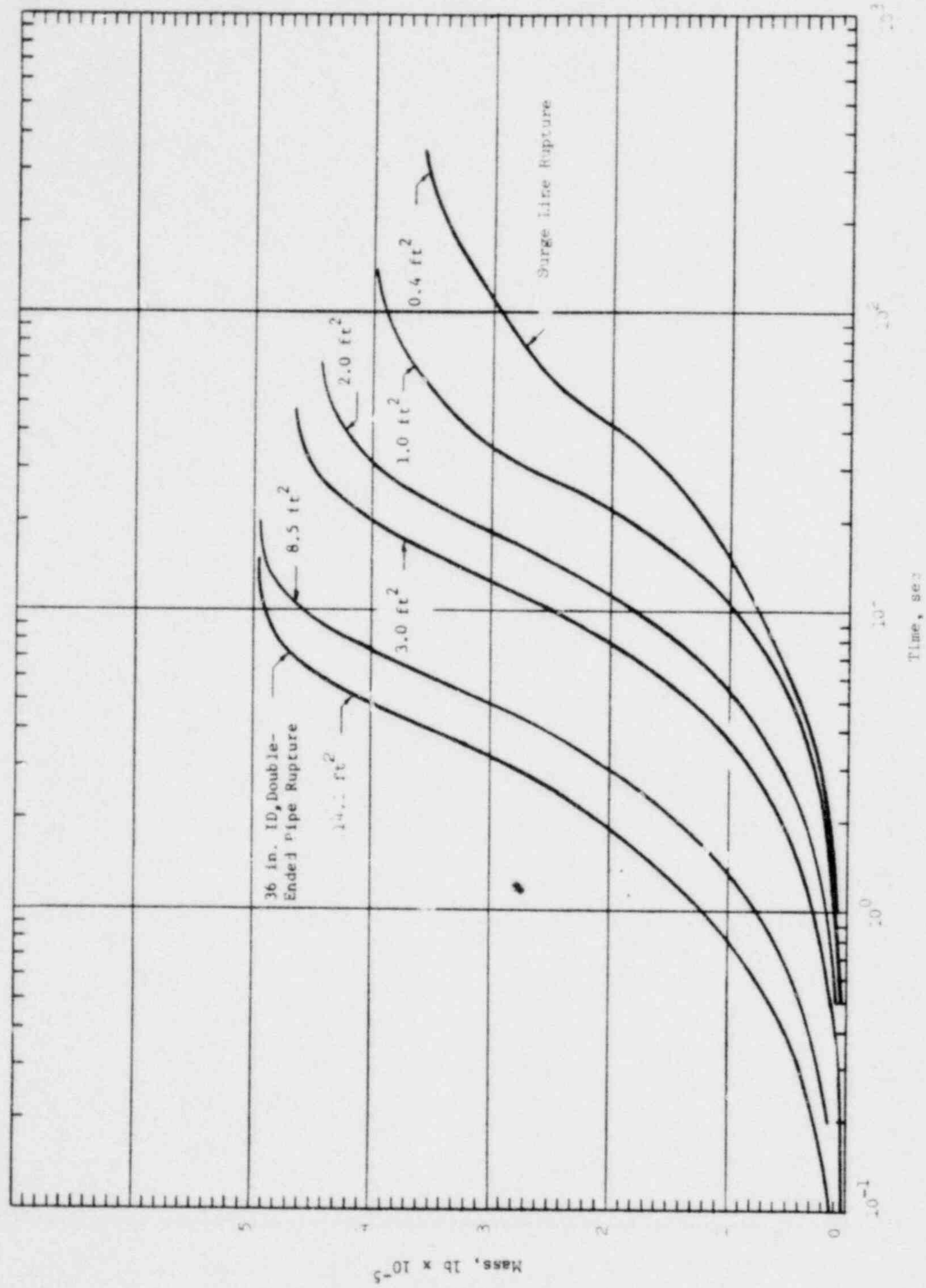


HOT SPOT CLAD TEMPERATURE VERSUS TIME FOR 36 IN. ID, DOUBLE-ENDED PIPE RUPTURE AND VARIABLE SINK TEMPERATURE

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-42



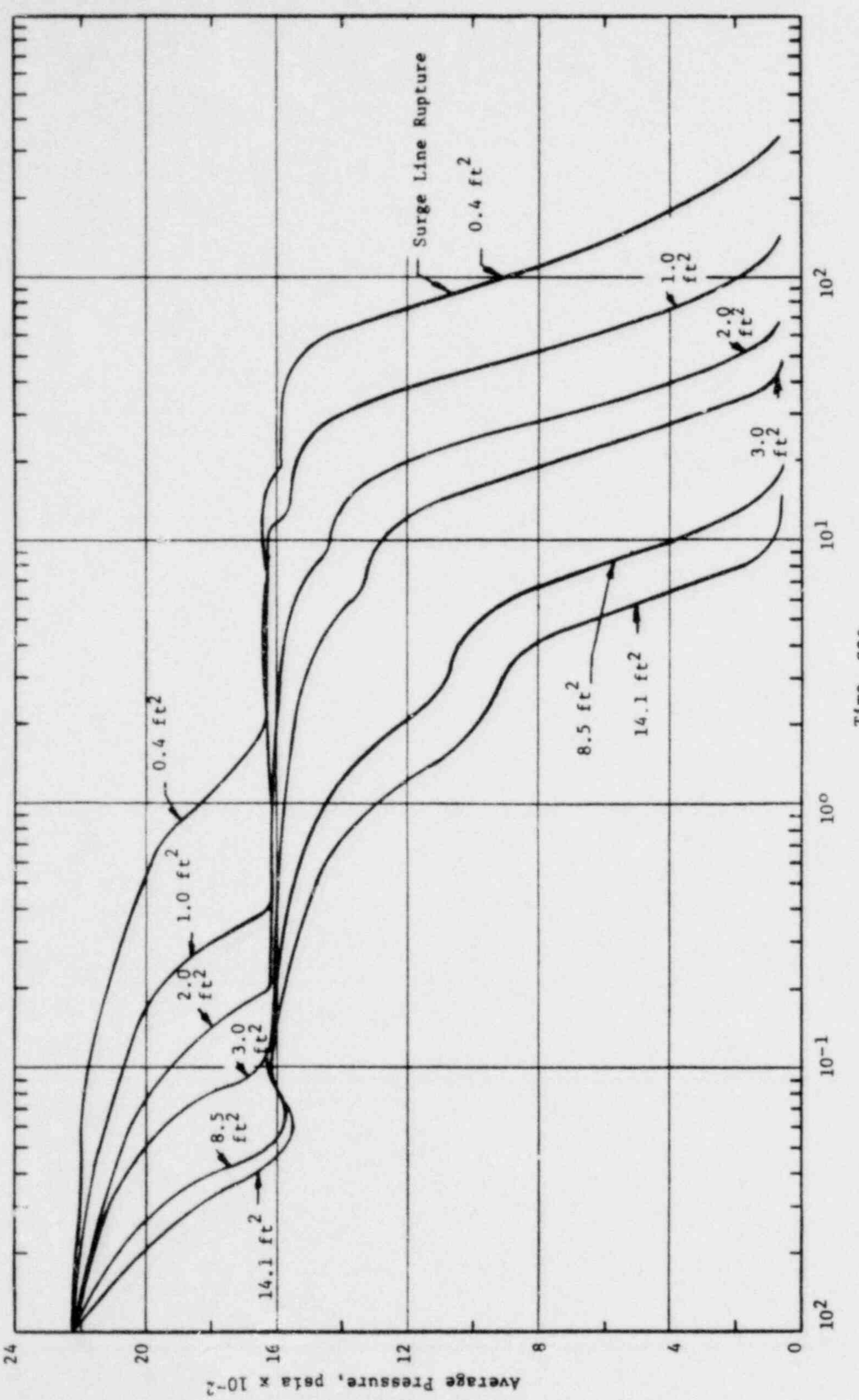
0118

MASS RELEASE TO REACTOR BUILDING FOR THE SPECTRUM OF HOT LEG RUPTURES

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-43

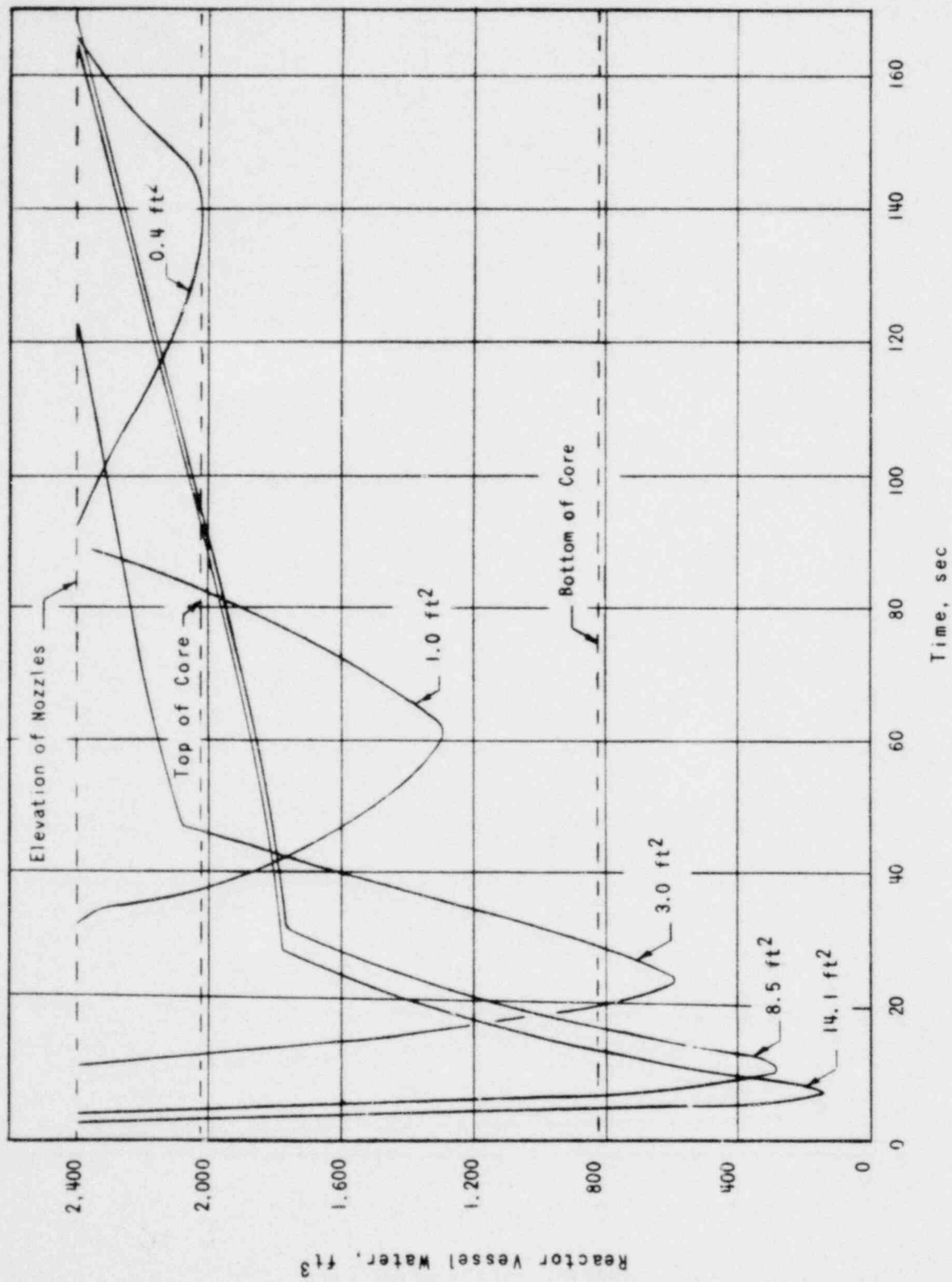


REACTOR COOLANT AVERAGE PRESSURE FOR THE SPECTRUM OF HOT LEG RUPTURES

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-44



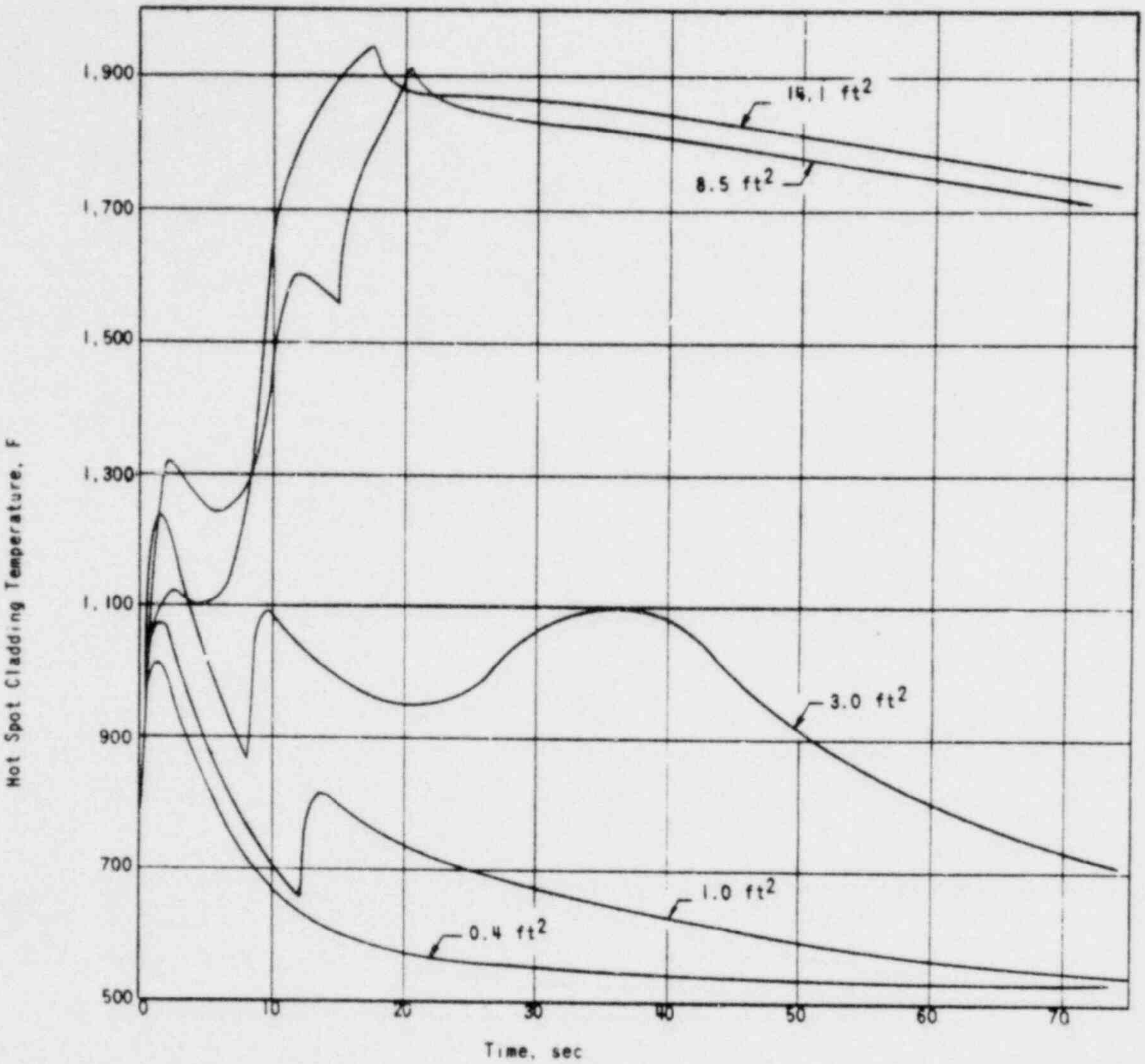
HOT LEG RUPTURES - REACTOR VESSEL WATER VOLUME VERSUS TIME INCLUDING EFFECTS OF BOILOFF AND INJECTION

CRYSTAL RIVER UNIT 3

0120

FLORIDA POWER CORPORATION FIGURE 14-44-a

AMEND. 5 (4-8-68)



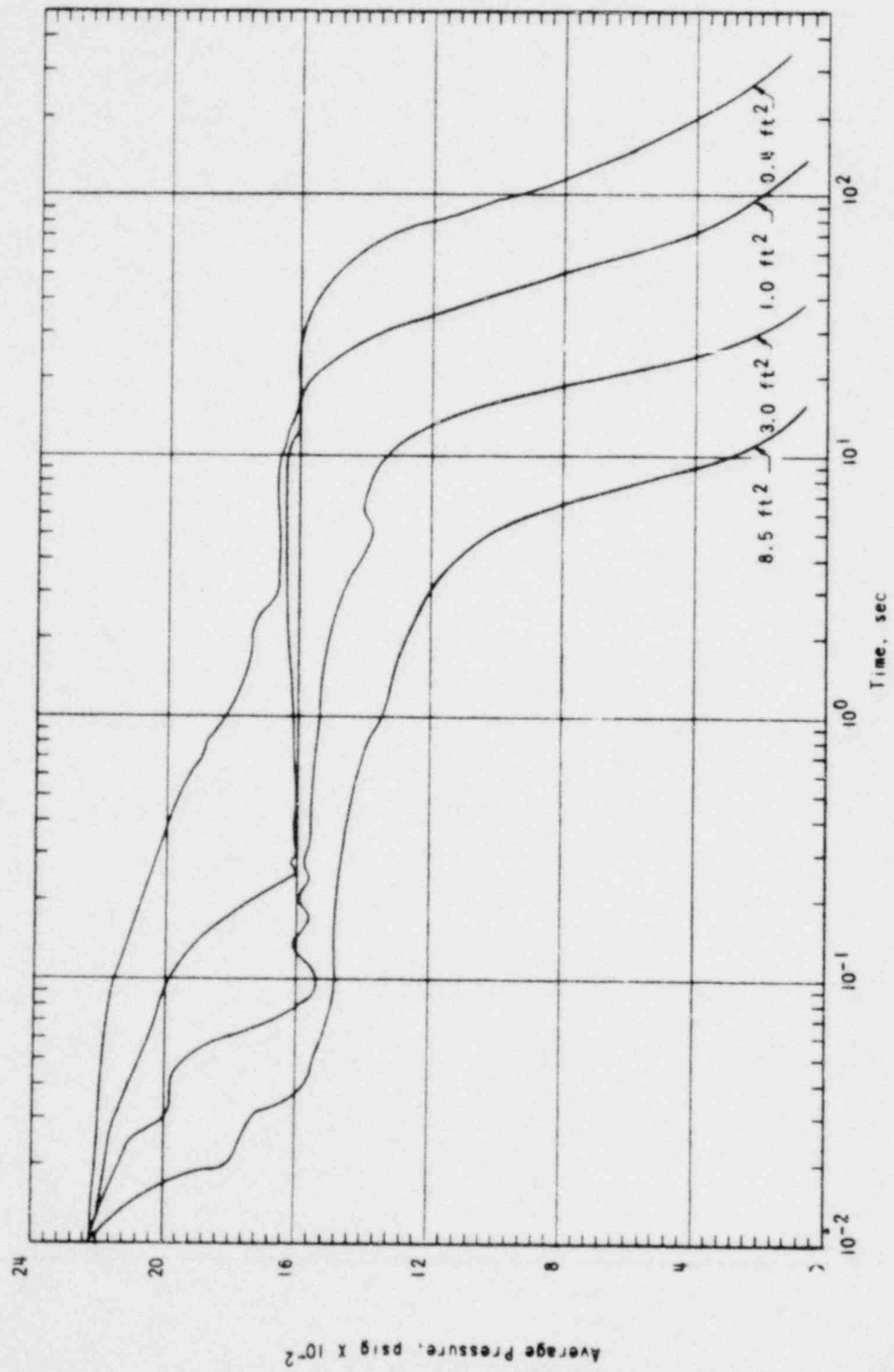
HOT SPOT CLADDING TEMPERATURE VERSUS
TIME FOR SPECTRUM OF HOT LEG RUPTURES

CRYSTAL RIVER UNIT 3

FLORIDA POWER CORPORATION FIGURE 14-44-b

AMEND. 5 (4-8-68)

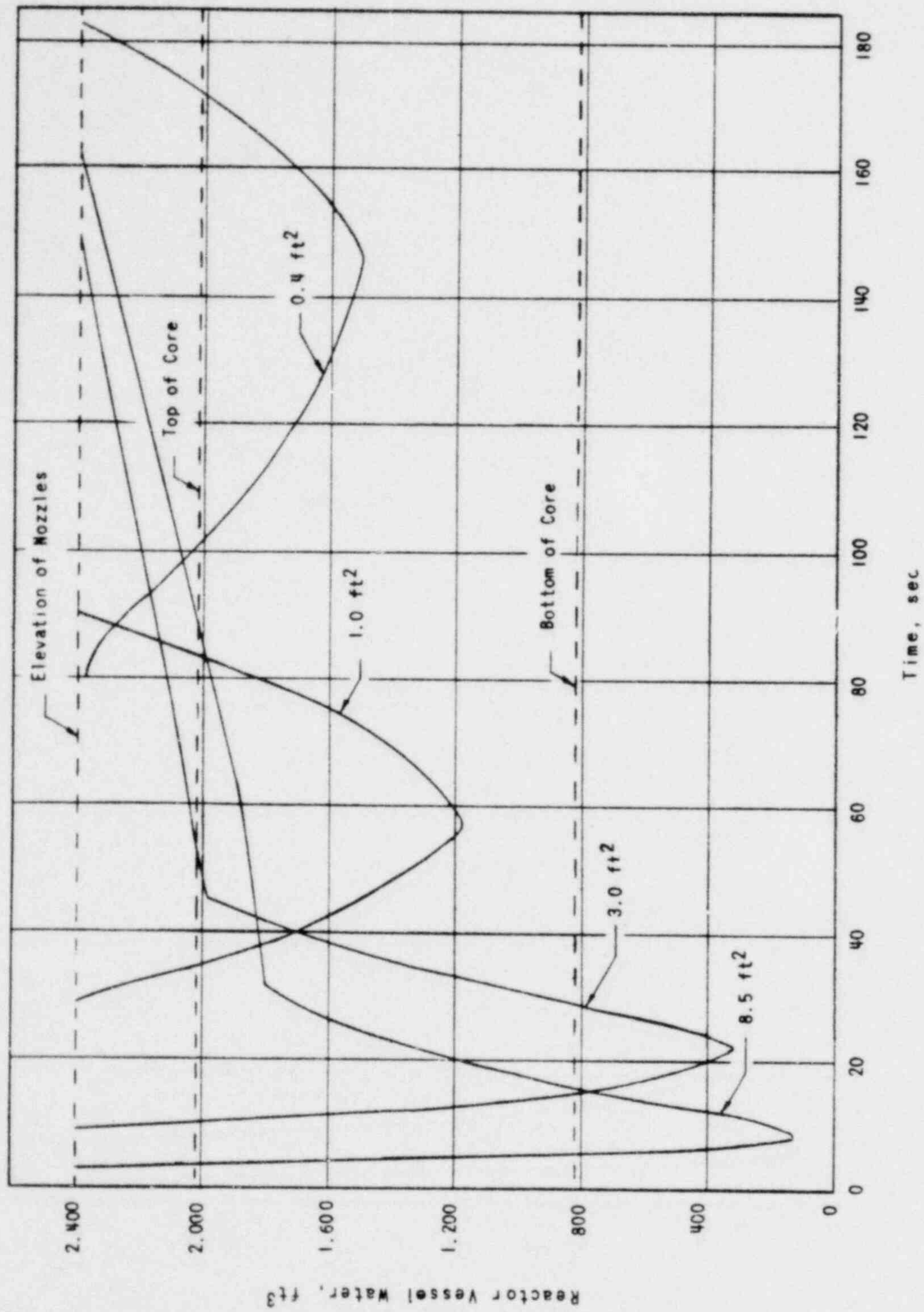
0121



REACTOR COOLANT AVERAGE PRESSURE
SPECTRUM OF COLD LEG RUPTURE SIZES
CRYSTAL RIVER UNIT 3

FLORIDA POWER CORPORATION FIGURE 14-44-c

AMEND. 5 (4-8-68)



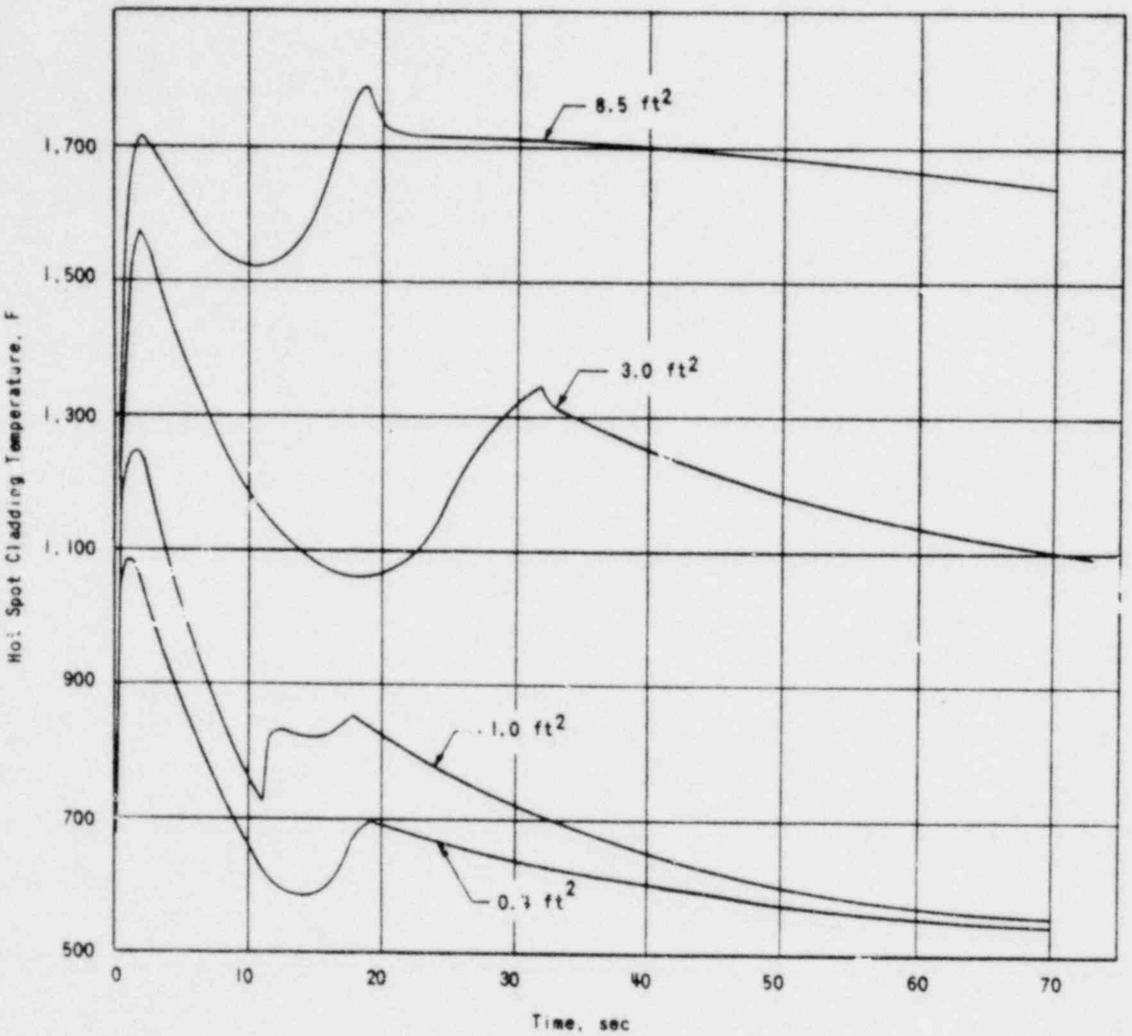
COLD LEG RUPTURES - REACTOR VESSEL WATER VOLUME VERSUS TIME INCLUDING EFFECTS OF BOILOFF AND INJECTION

CRYSTAL RIVER UNIT 3

FLORIDA POWER CORPORATION FIGURE 14-44-d

AMEND. 5 (4-8-68)

0123



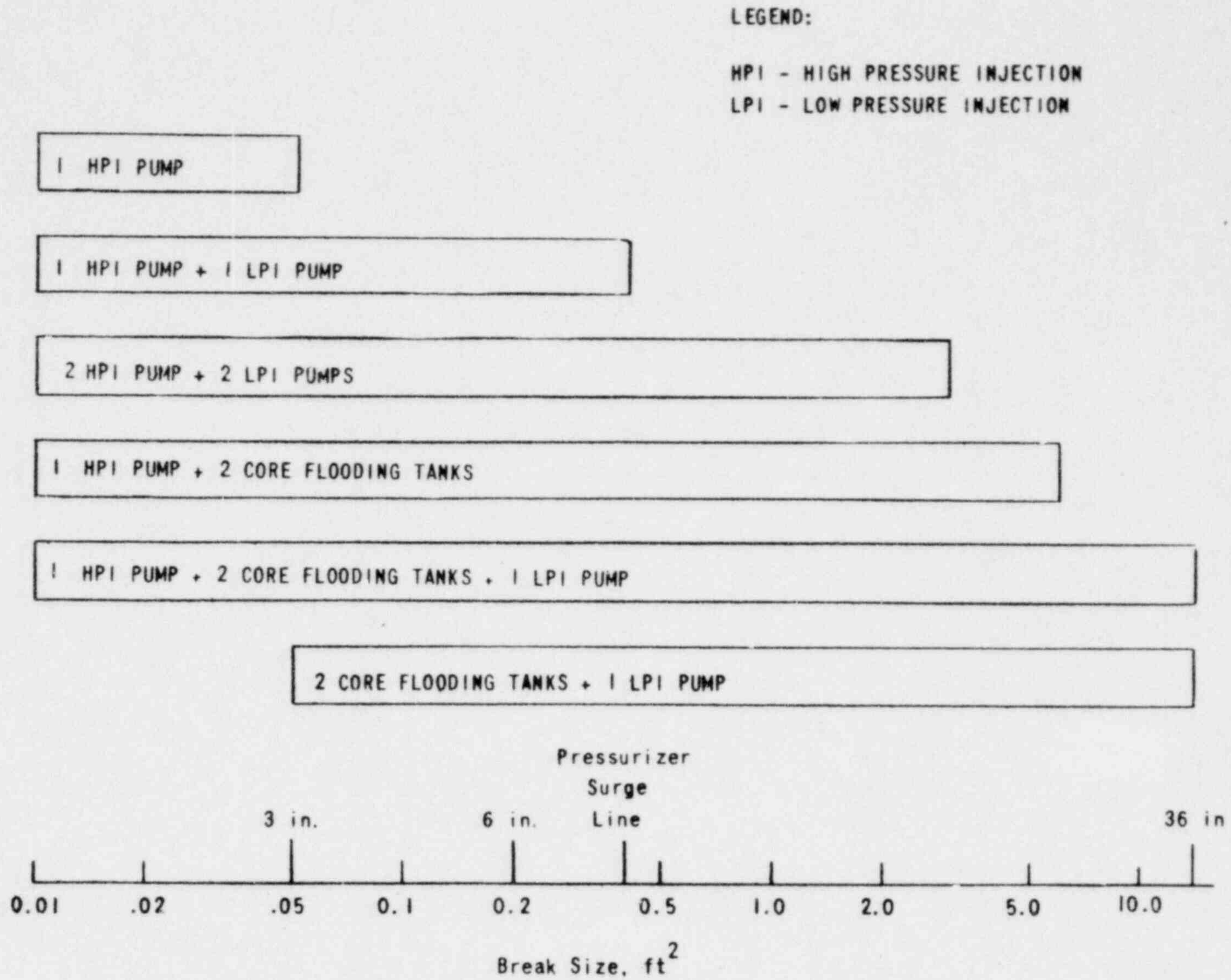
HOT SPOT CLADDING TEMPERATURE VERSUS TIME FOR SPECTRUM OF COLD LEG RUPTURES

CRYSTAL RIVER UNIT 3

0124

FLORIDA POWER CORPORATION FIGURE 14-44-e

AMEND. 5 (4-8-68)



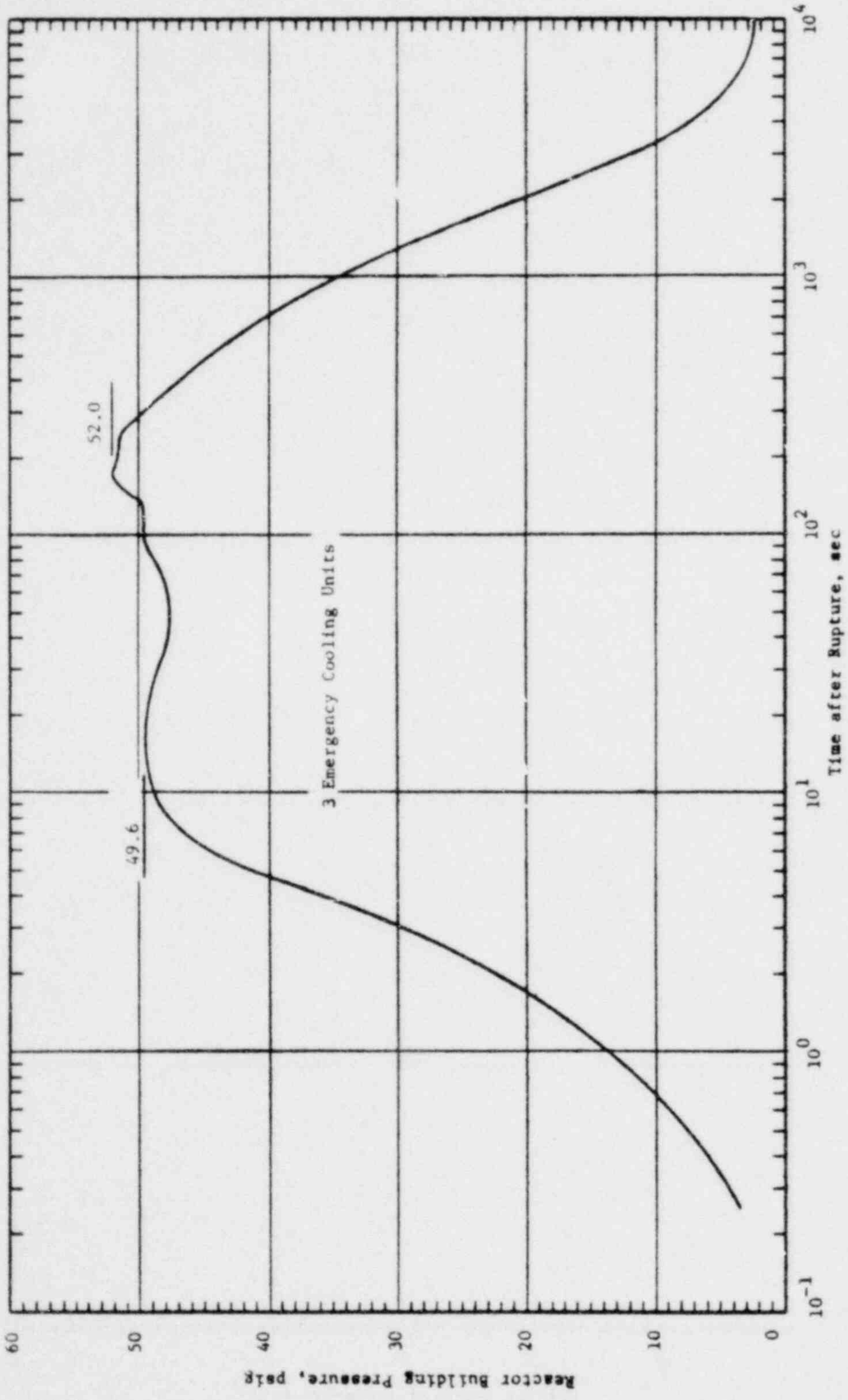
EMERGENCY CORE COOLING SYSTEMS CAPABILITY

CRYSTAL RIVER UNIT 3

FIGURE 14-44-f

AMEND. 5 (4-8-68)

0125



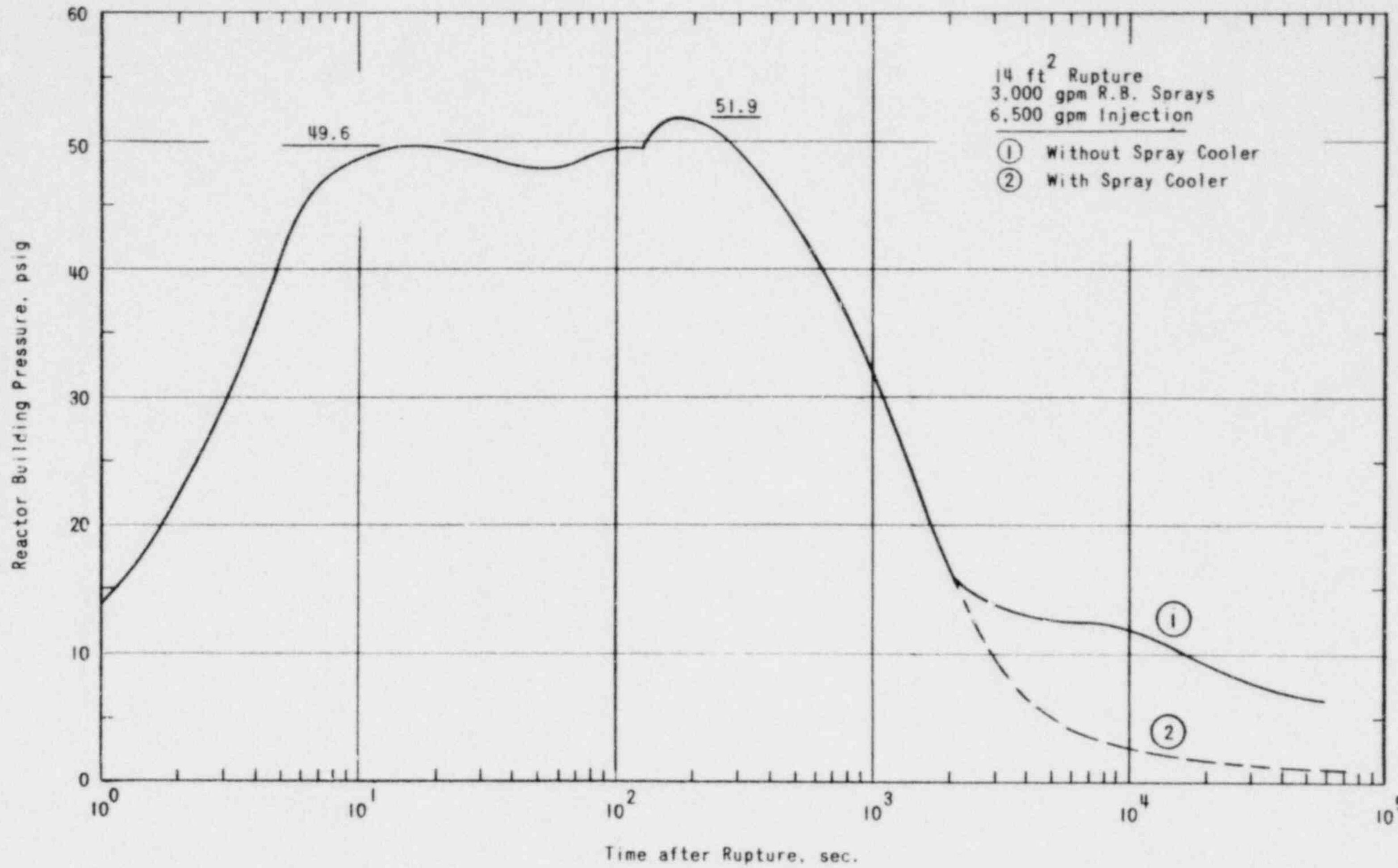
0126

REACTOR BUILDING PRESSURE VERSUS TIME
36 IN. ID, DOUBLE - ENDED RUPTURE

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-45

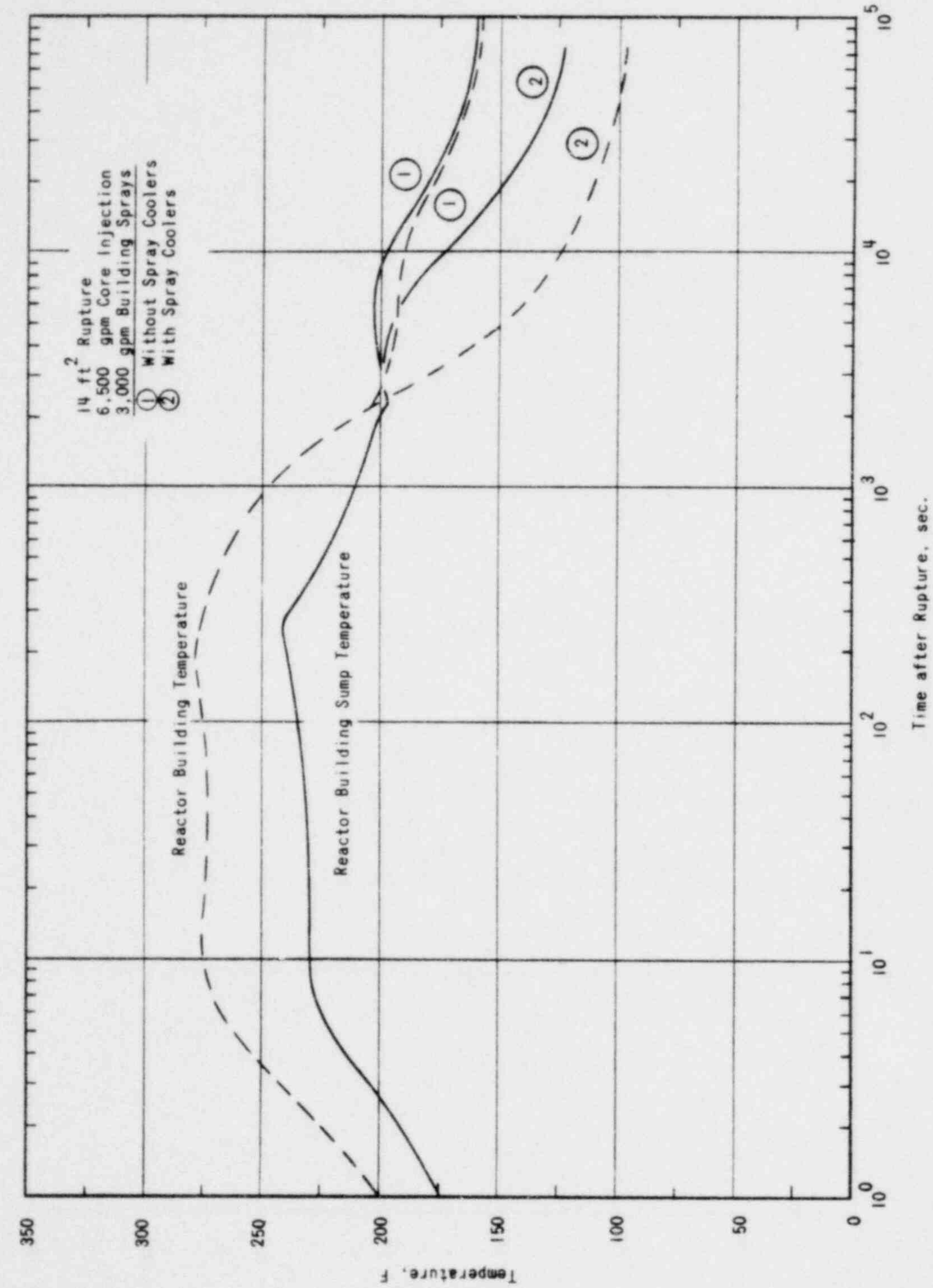


REACTOR BUILDING PRESSURE VS TIME FOR A 36 IN. I.D. DOUBLE-ENDED RUPTURE WITH AND WITHOUT COOLING OF THE RECIRCULATED SPRAY WATER

CRYSTAL RIVER UNITS 3 & 4

FIGURE 14-46

0127



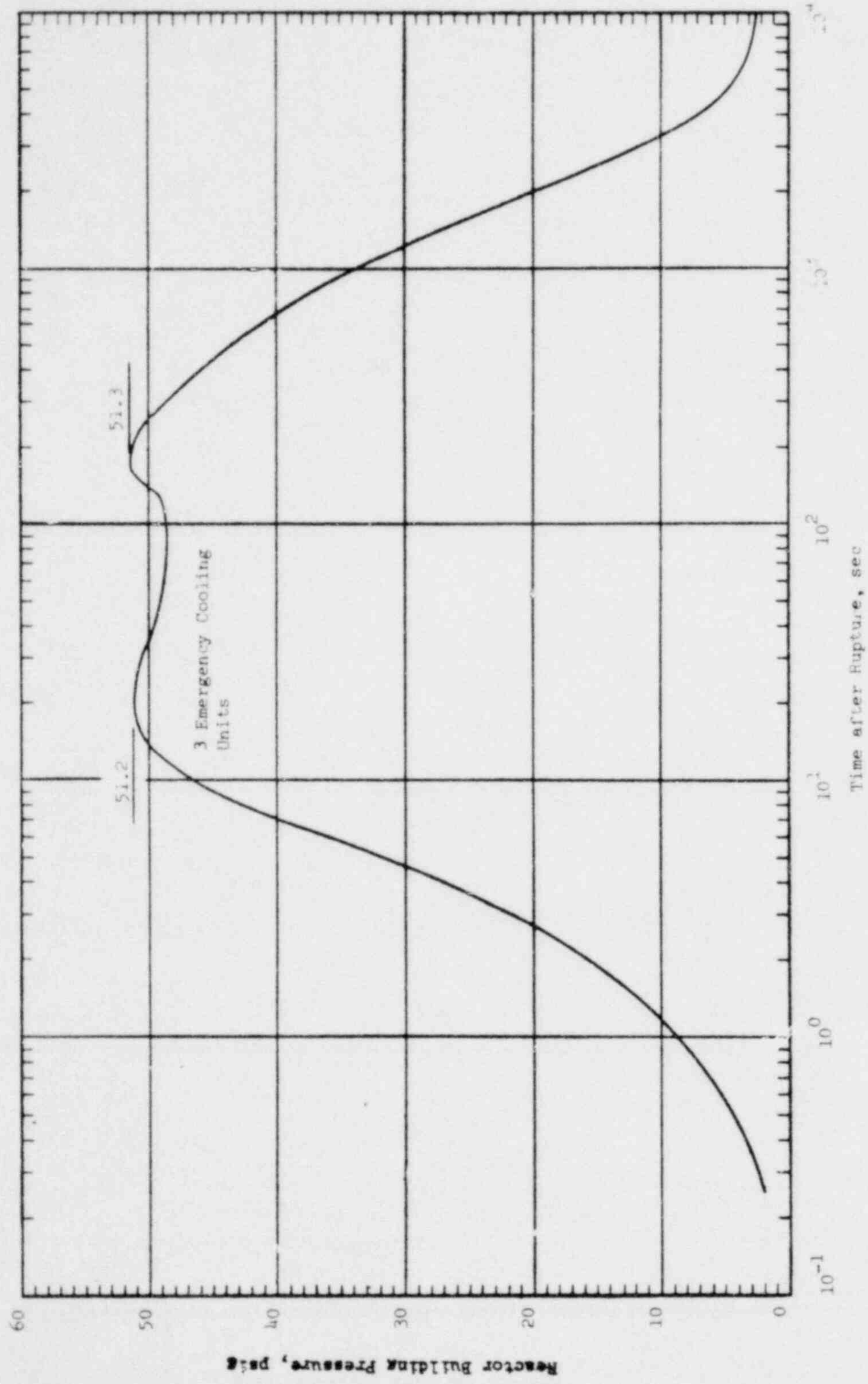
REACTOR BUILDING ATMOSPHERE AND SUMP
COOLANT TEMPERATURES FOLLOWING A 36 IN.
I.D. DOUBLE-ENDED RUPTURE

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-47

0128

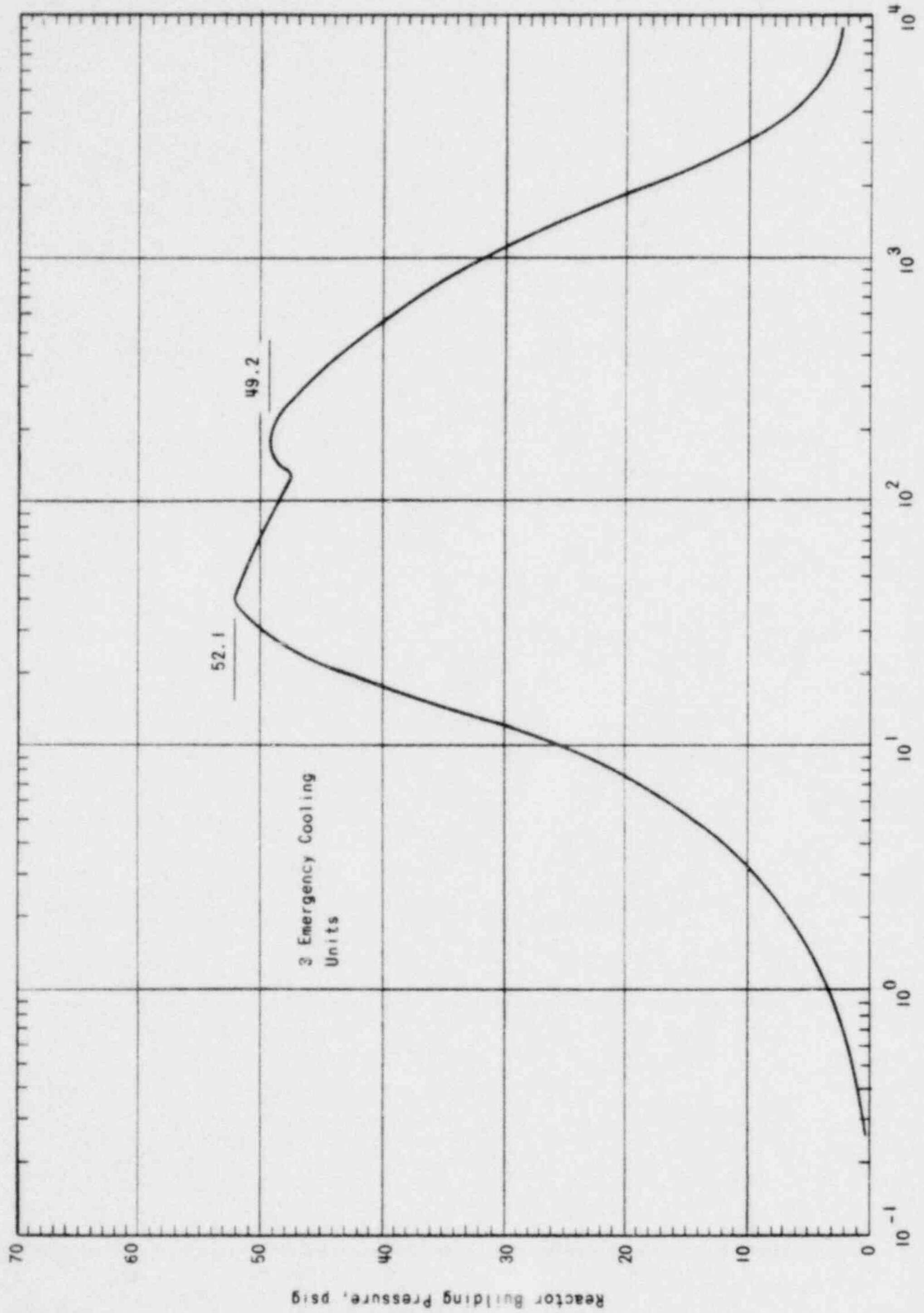


REACTOR BUILDING PRESSURE VERSUS TIME
AFTER RUPTURE - 8.5 FT^2 RUPTURE

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION FIGURE 14-48

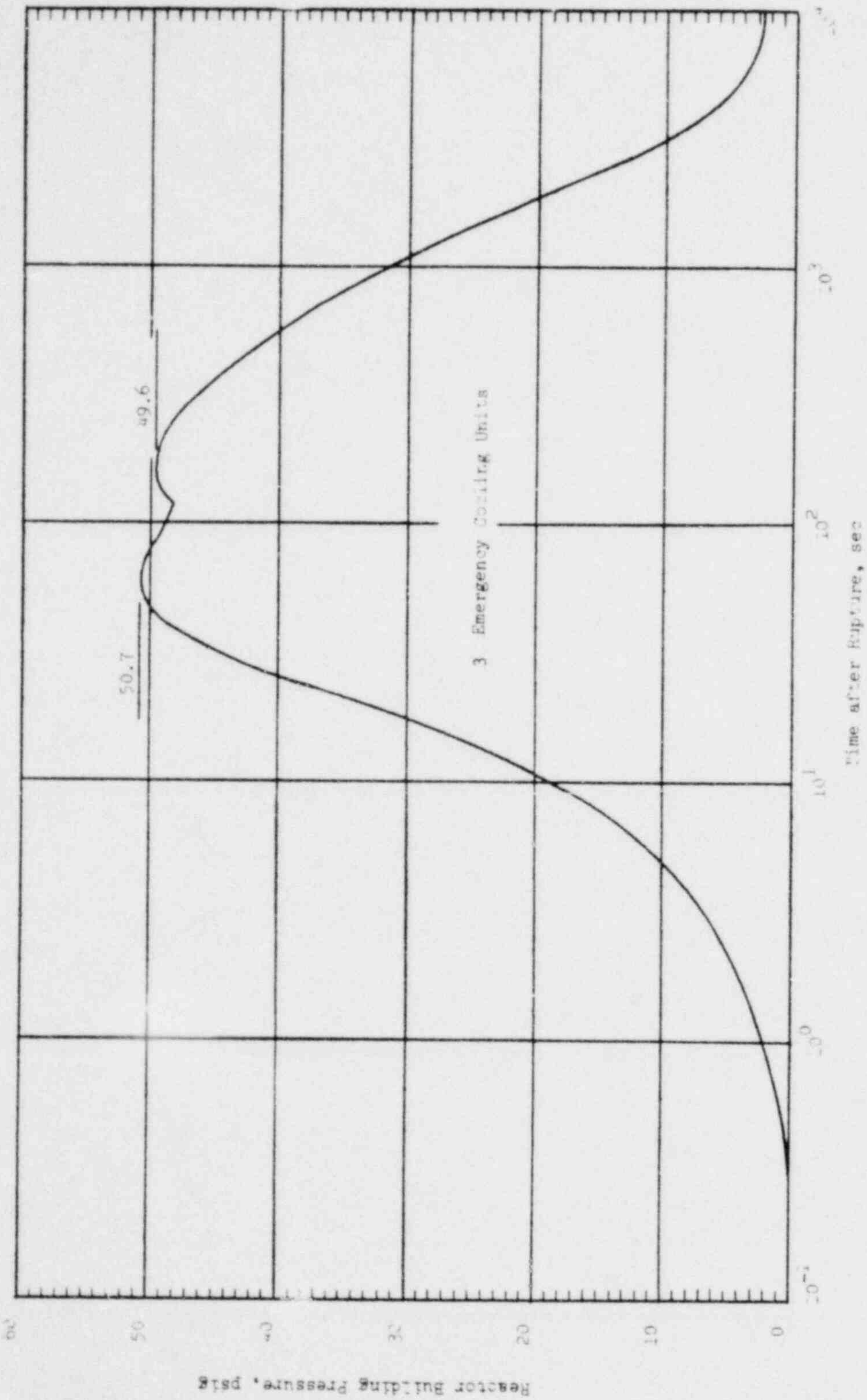
0129



REACTOR BUILDING PRESSURE VERSUS TIME AFTER RUPTURE - 3 FT^2 RUPTURE

CRYSTAL RIVER UNITS 3 & 4

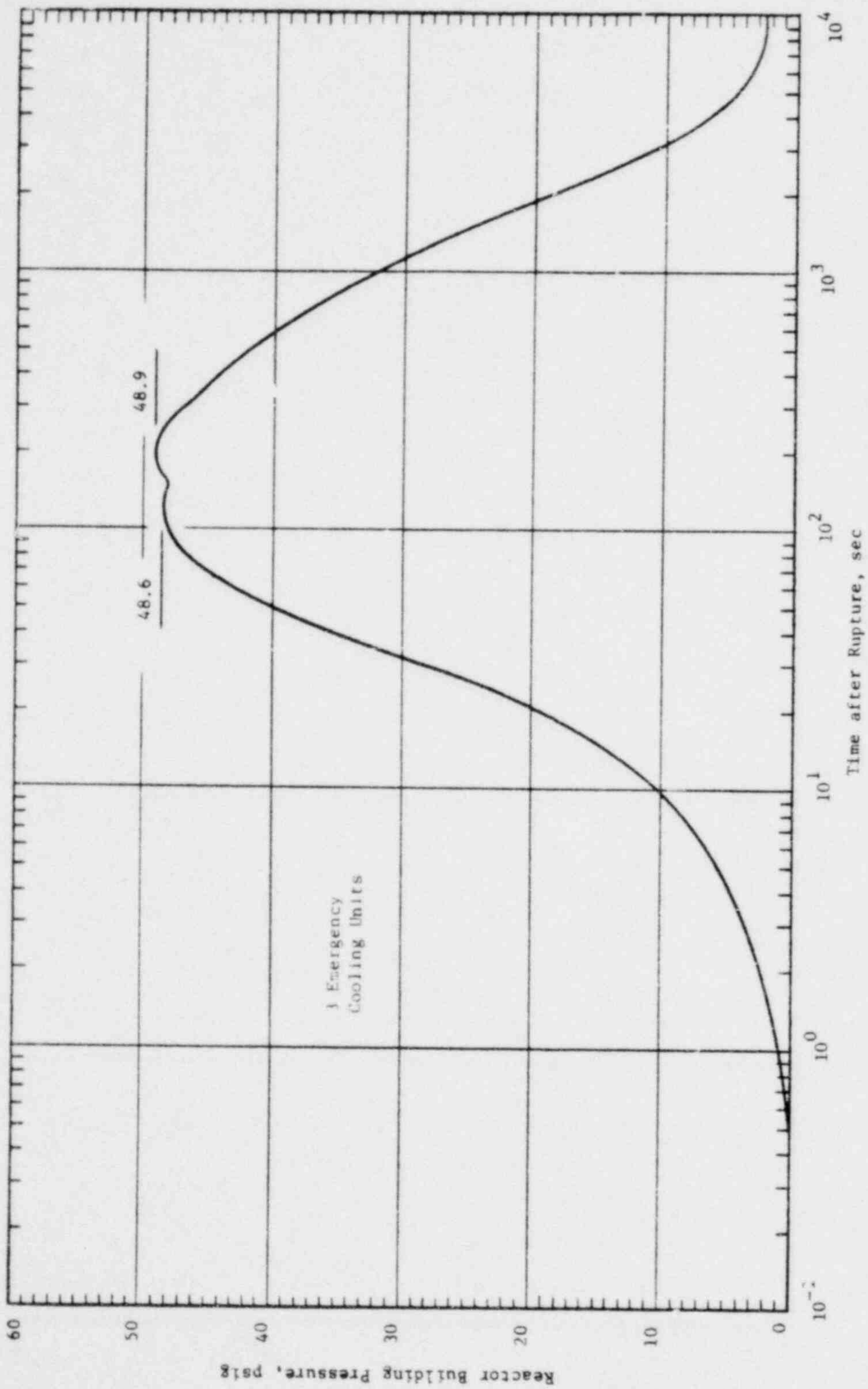
FIGURE 14-49



REACTOR BUILDING PRESSURE VERSUS TIME AFTER RUPTURE - 2 FT^2 RUPTURE

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION FIGURE 14-50



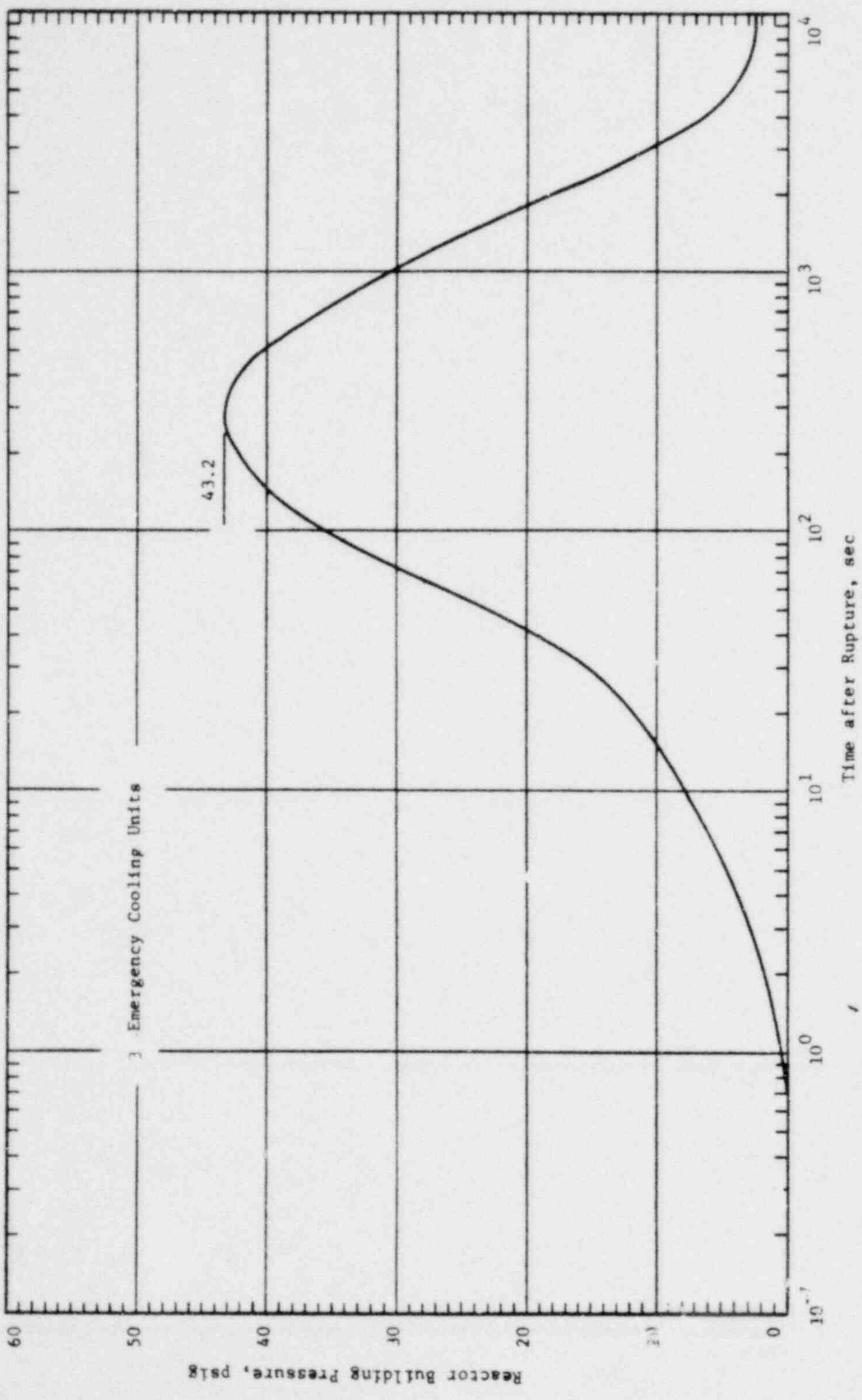
REACTOR BUILDING PRESSURE VERSUS TIME AFTER RUPTURE - 1 FT^2 RUPTURE

CRYSTAL RIVER UNITS 3 & 4

0132

FLORIDA POWER CORPORATION

FIGURE 14-51



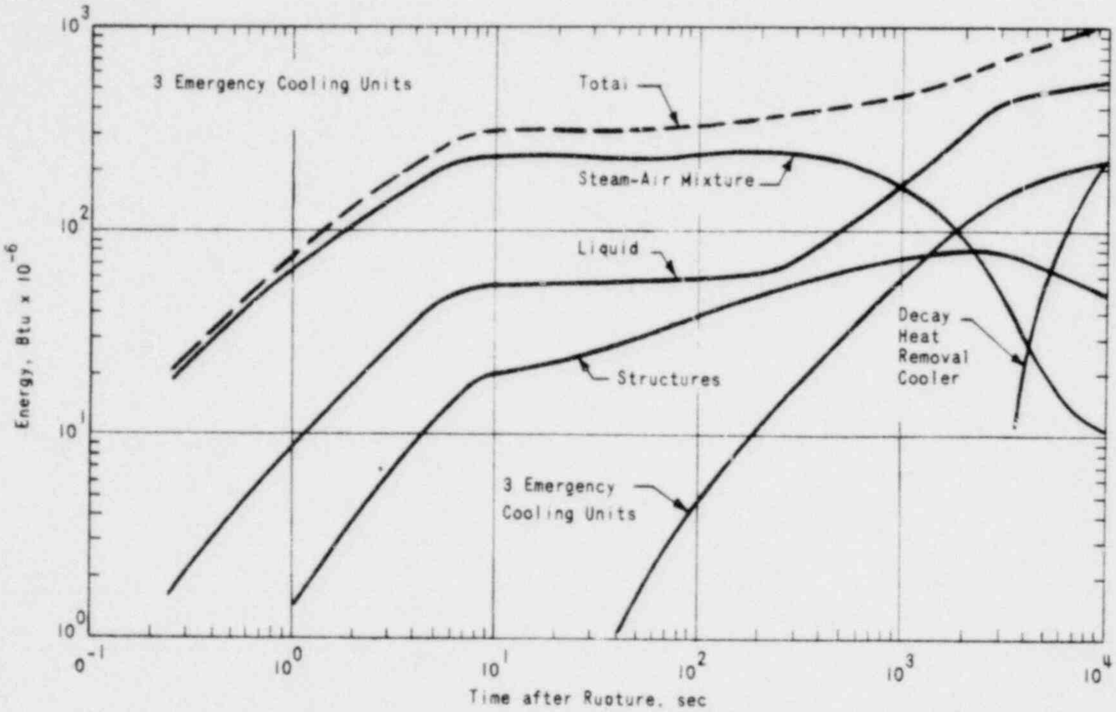
0133

REACTOR BUILDING PRESSURE VERSUS TIME AFTER RUPTURE - 0.4 FT^2 RUPTURE

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-52



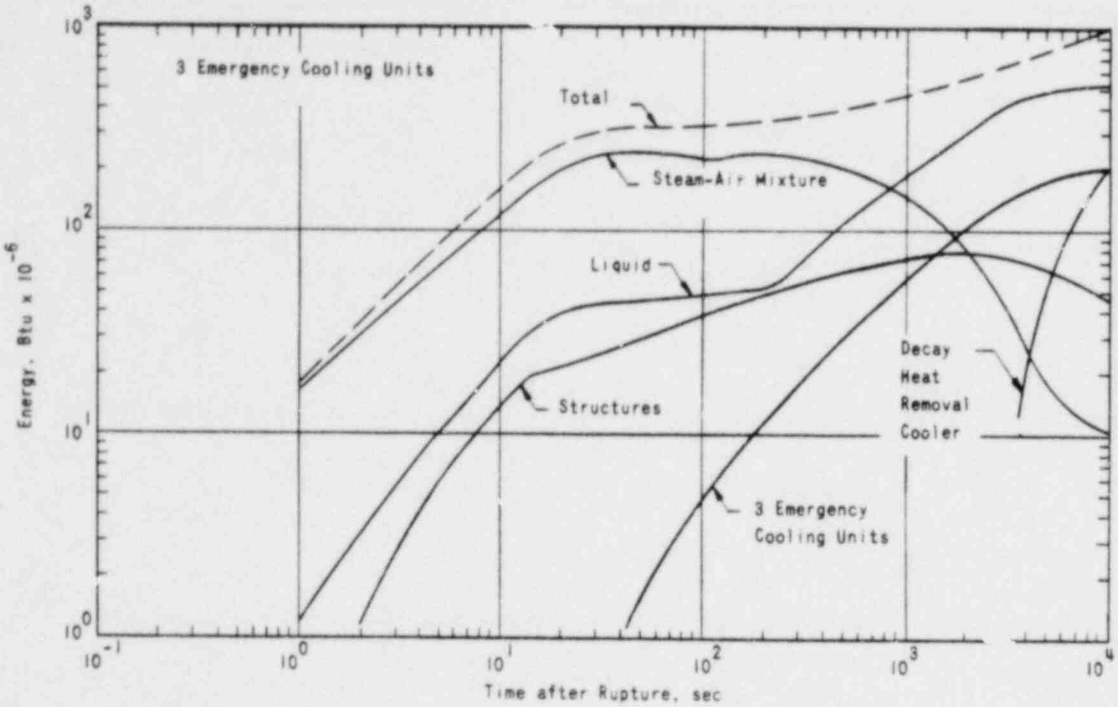
0134

REACTOR BUILDING ENERGY INVENTORY FOR 36 IN. ID, DOUBLE-ENDED RUPTURE

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-53



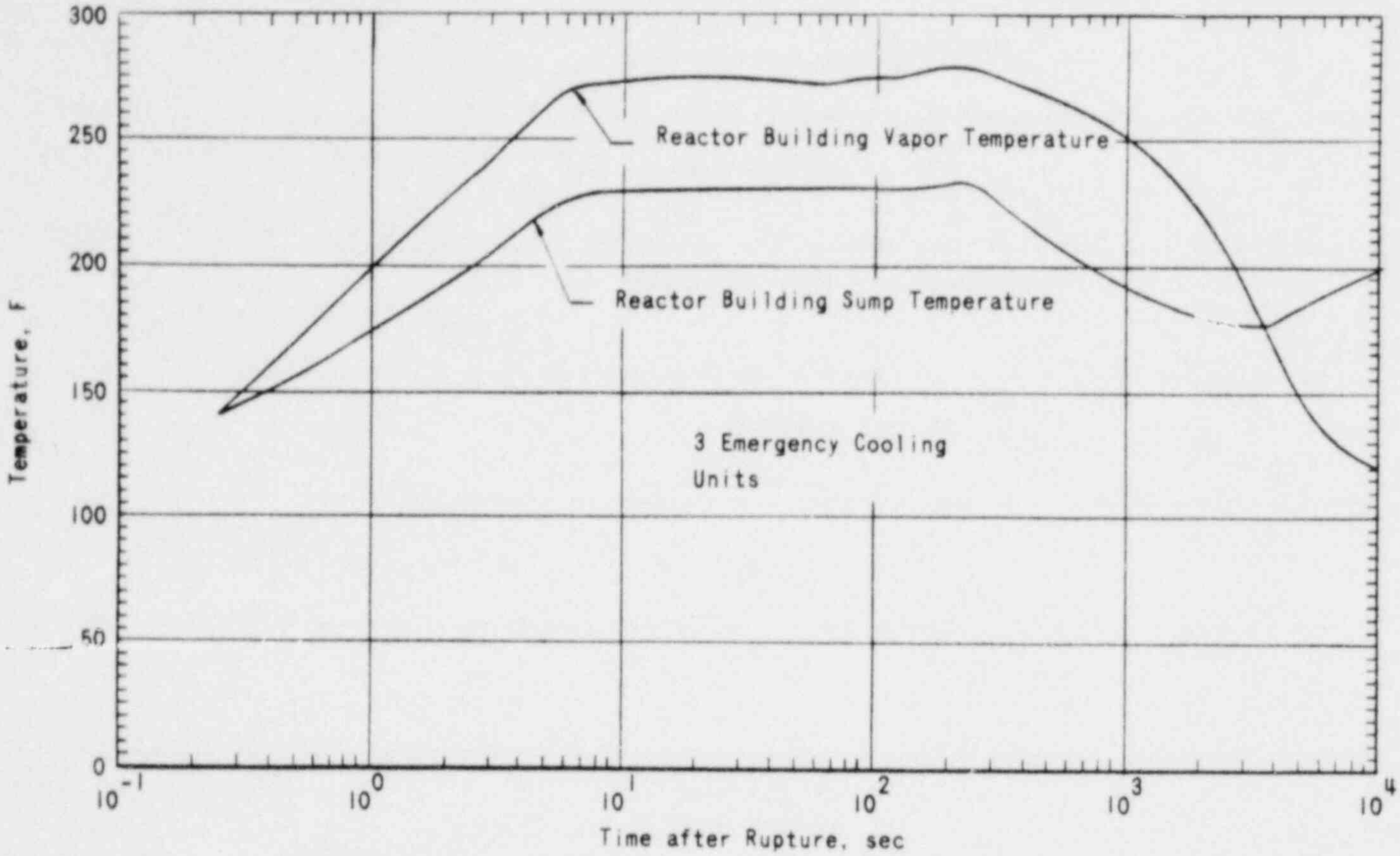
REACTOR BUILDING ENERGY INVENTORY
FOR 3 FT^2 RUPTURE

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-54

0135



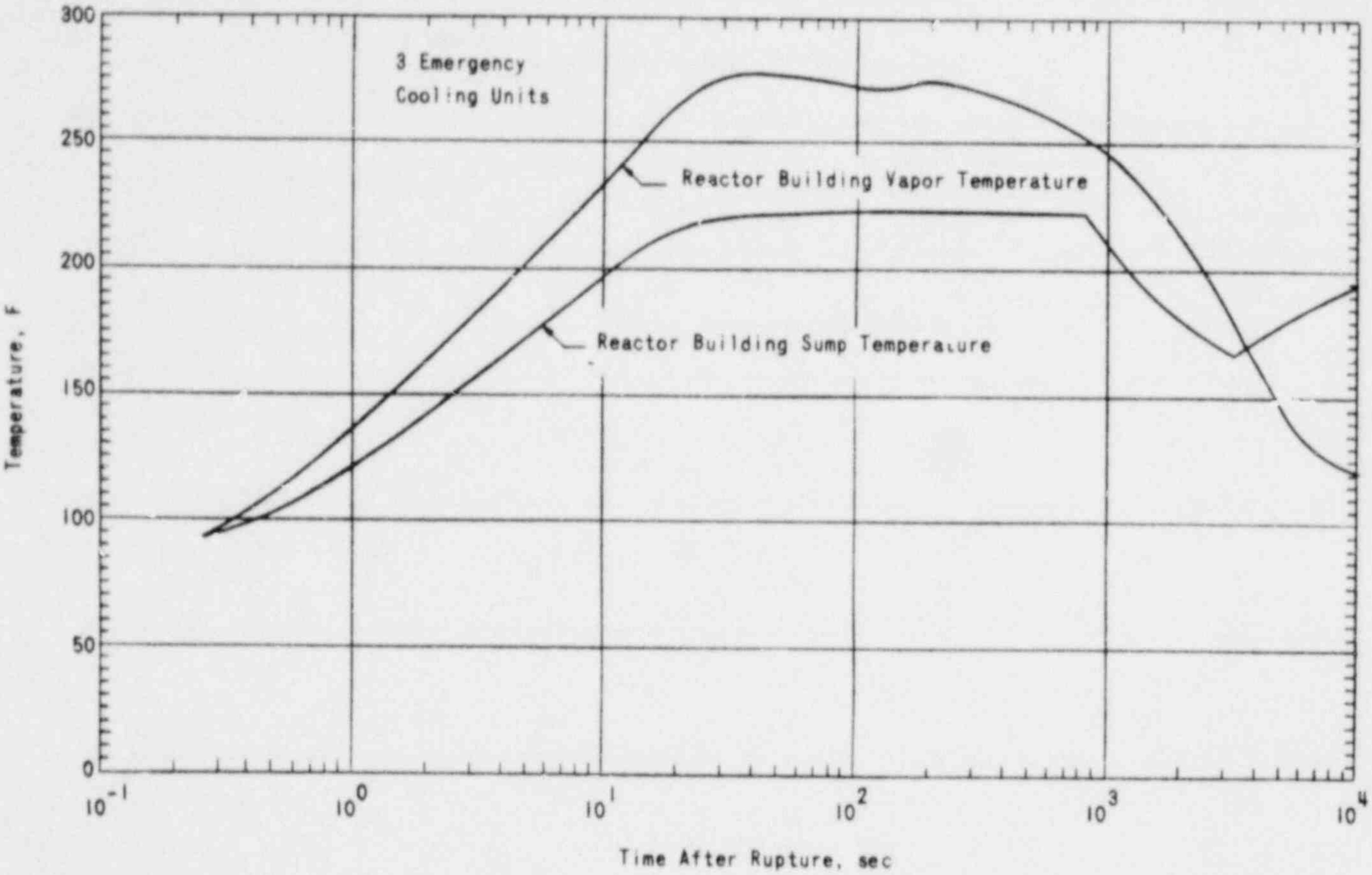
REACTOR BUILDING VAPOR AND SUMP TEMPERATURES FOR 36 IN. ID, DOUBLE-ENDED RUPTURE AS A FUNCTION OF TIME AFTER THE RUPTURE

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-55

0136

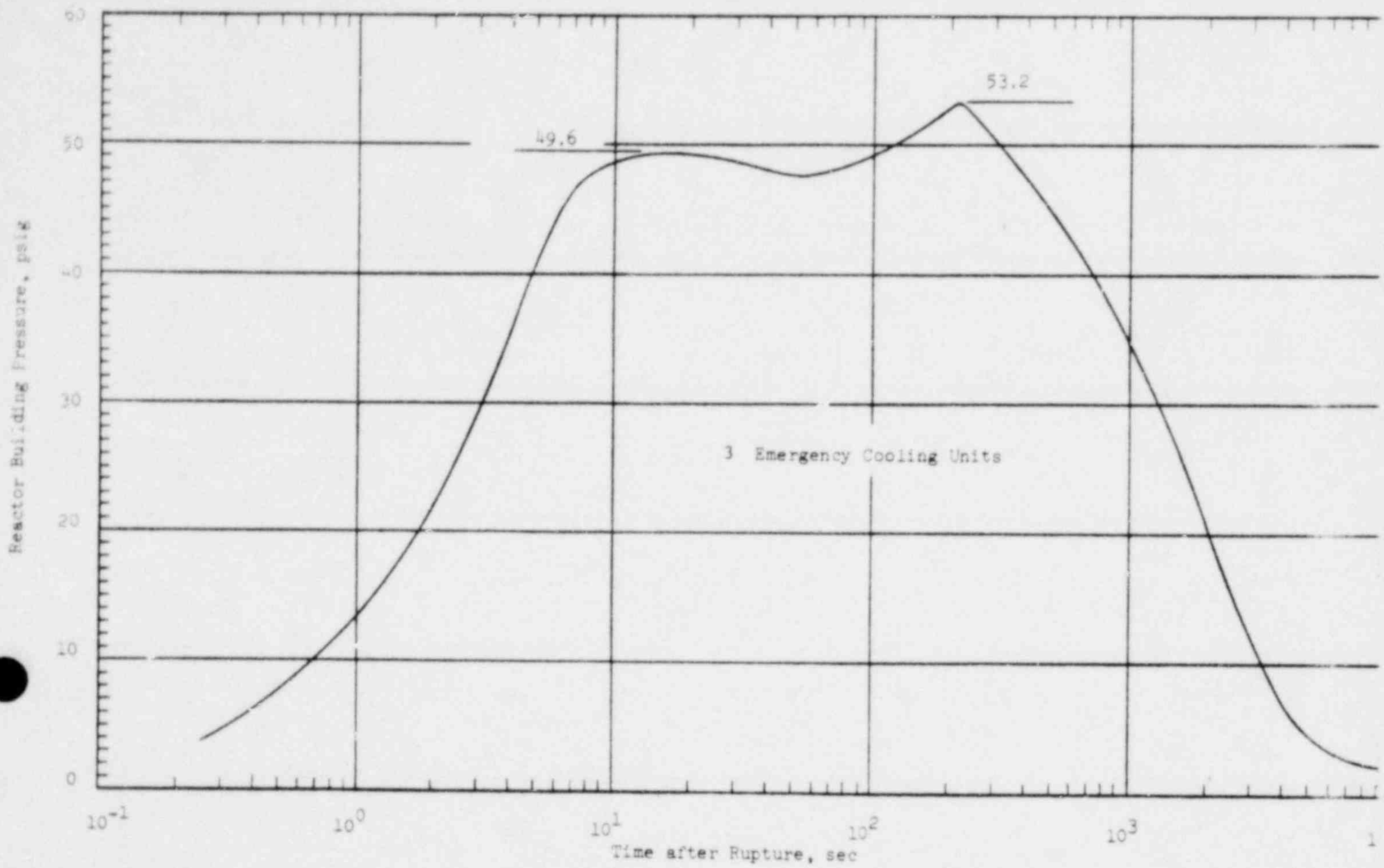


REACTOR BUILDING VAPOR AND SUMP TEMPERATURES
AS A FUNCTION OF TIME AFTER RUPTURE
3 FT^2 RUPTURE

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION FIGURE 14-56

0137



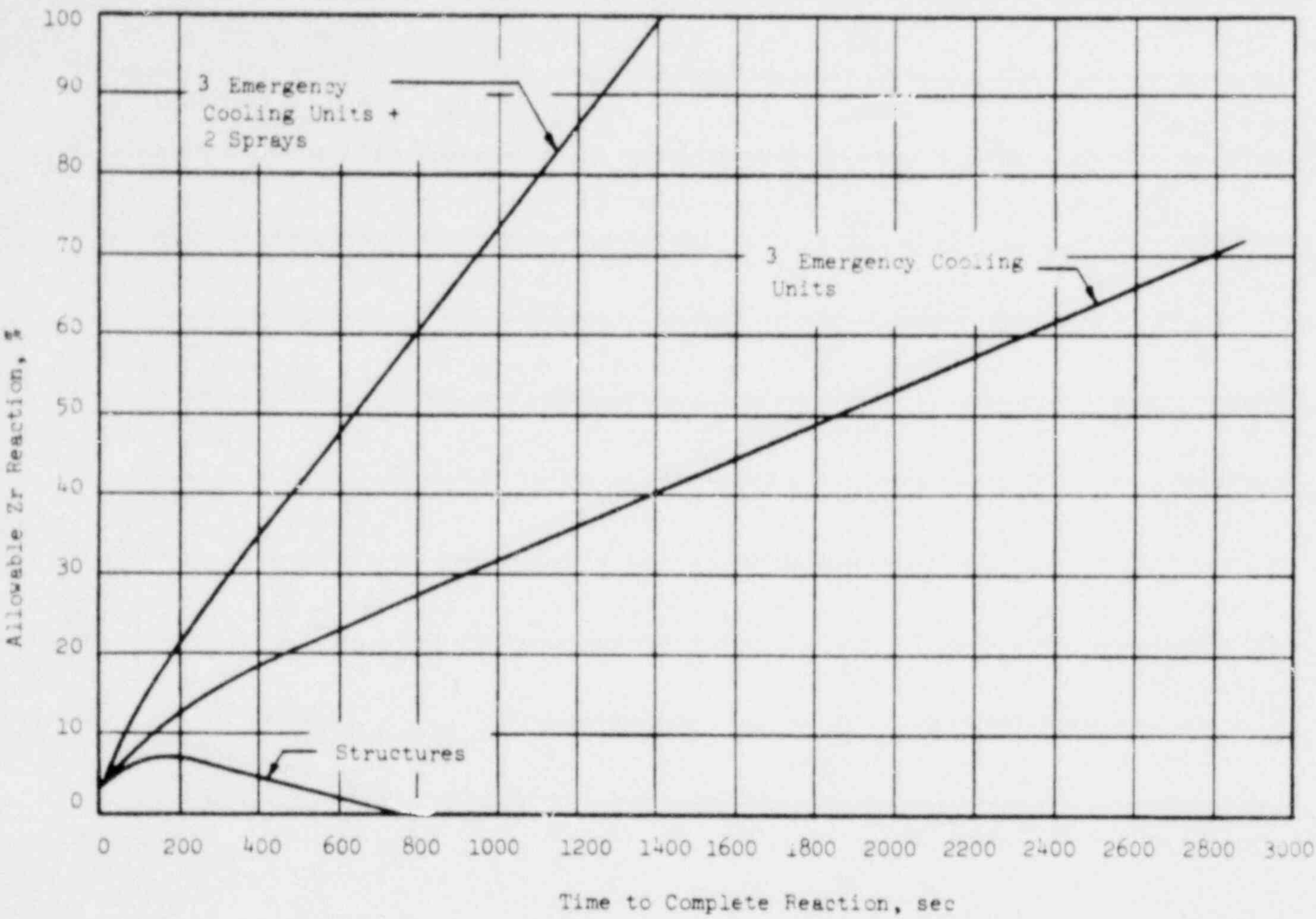
0138

CRITERION 17 CASE FOR 36 IN. ID, DOUBLE-ENDED RUPTURE

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-57



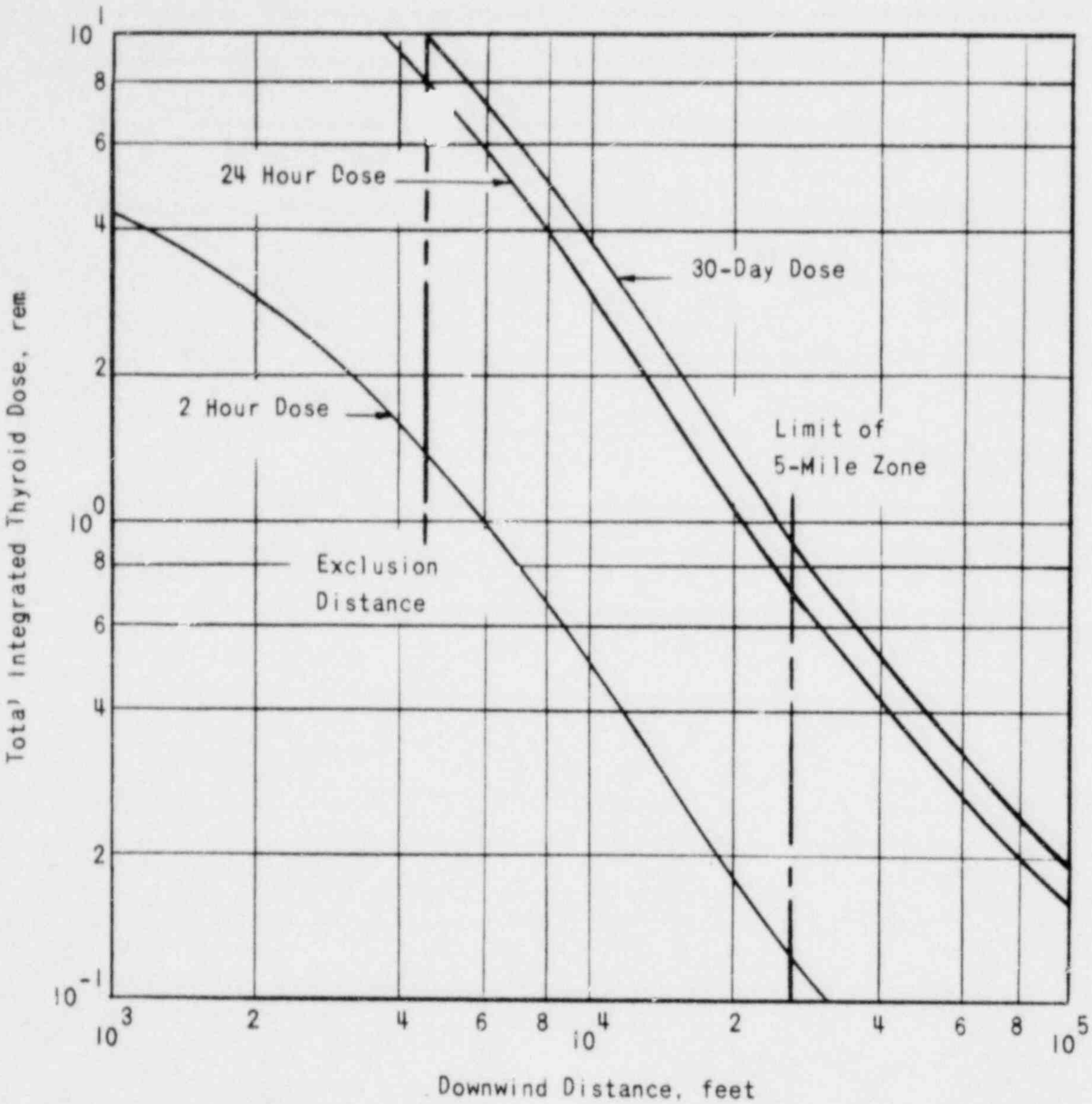
REACTOR BUILDING ZR REACTION CAPABILITY
FOR 55 PSIG DESIGN PRESSURE

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-5?

0139



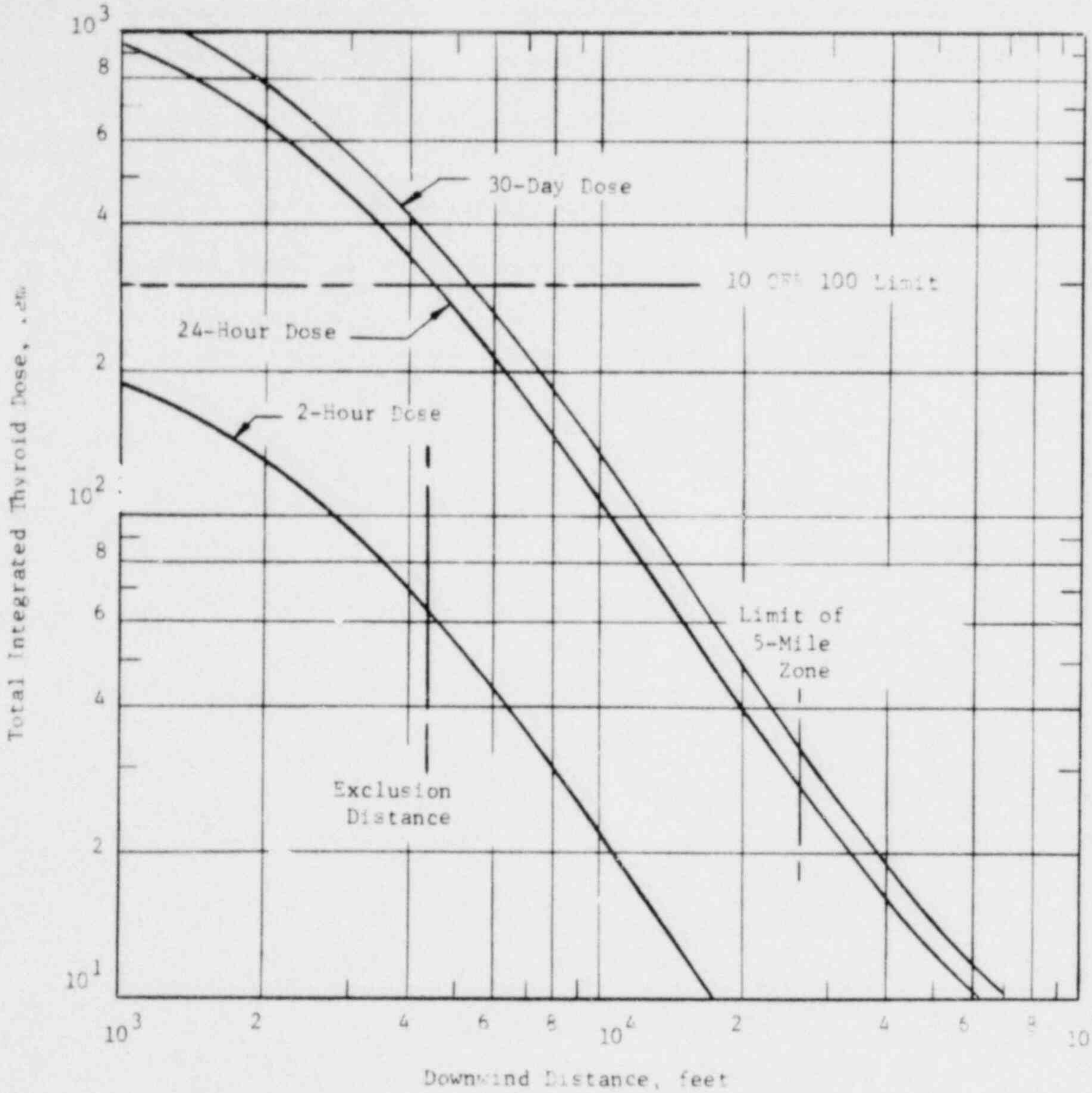
0140

THYROID DOSE FROM LOSS OF COOLANT COOLANT ACCIDENT - 2 HOURS, 24 HOURS, AND 30 DAYS DOSES

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-59



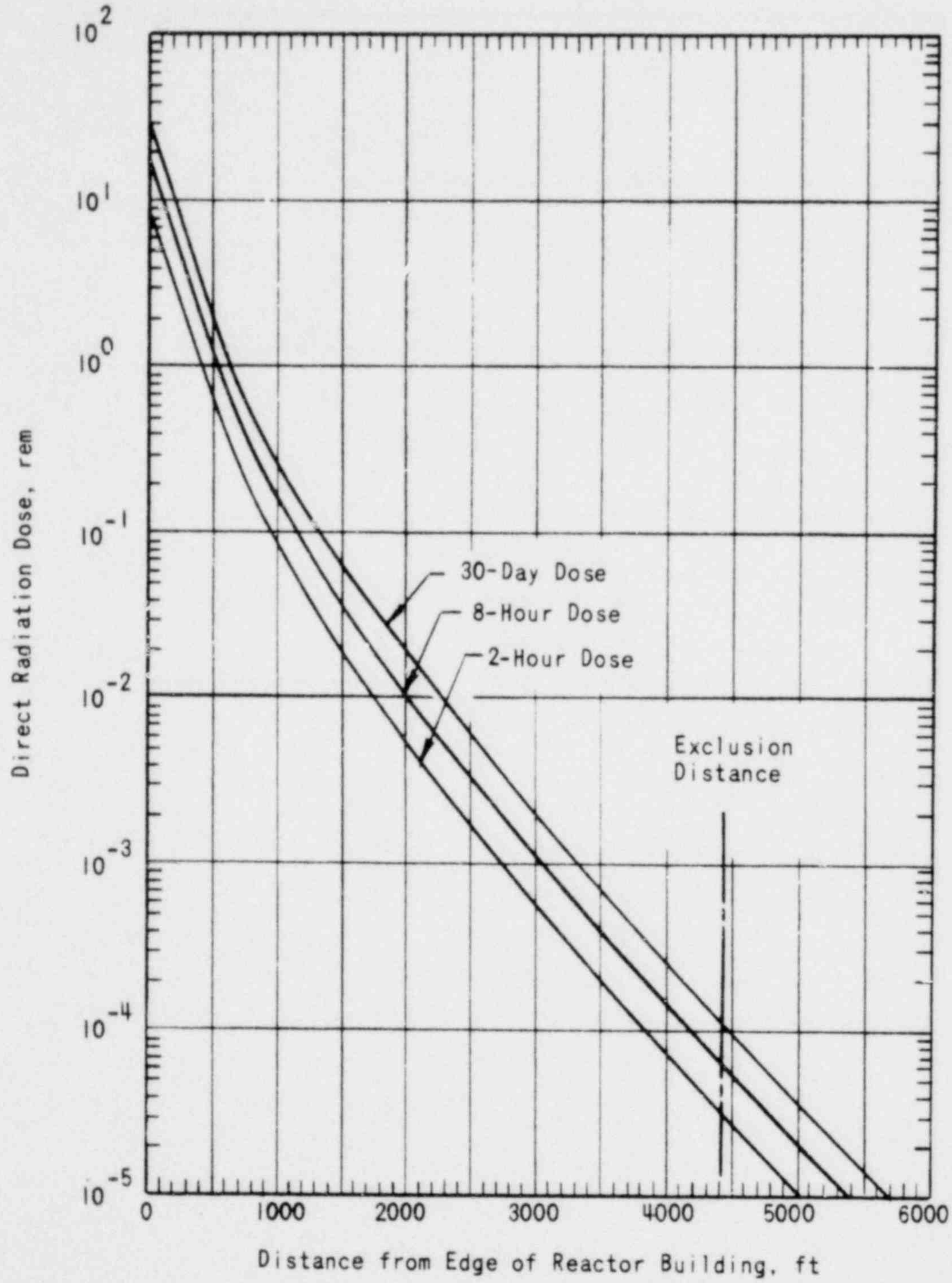
0141

MAXIMUM HYPOTHETICAL ACCIDENT
THYROID DOSE ASSUMING FISSION
PRODUCT RELEASE PER TID - 14844

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-60



0142

INTEGRATED DIRECT DOSE FOLLOWING MHA WITH 3½ - FOOT REACTOR BUILDING WALL THICKNESS

CRYSTAL RIVER UNITS 3 & 4

FLORIDA POWER CORPORATION

FIGURE 14-61