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VERMONT YANKEE NUCLEAR POWER STATION SINGLE LOOP OPERATION

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1. INTRODUCTION AND SUMMARY

The purpose of this report is to provide the justification for the operation of the Vermont Yankee Nuclear Power Station with one recirculation loop out of service. The current technical specifications limit single loop operation to a maximum 24-hour period (Technical Specification 3.6.G.1). The analysis described herein provides the basis for removing the 24-hour time constraint.

The capability of operating at reduced power with a single recirculation loop is highly desirable, from a plant availability/outage planning standpoint, in the event that maintenance of a recirculation pump or other component renders one loop inoperative. To justify single loop operation, the safety analyses documented in the Final Safety Analysis Report and Reference 1 were reviewed for one pump operation. Increased uncertainties in the core total flow and Traversing In-Core Probe (TIP) readings can result in a 0.01 incremental increase in the Minimum Critical Power Ratio (MCPR) fuel cladding integrity safety limit during single loop operation. This 0.01 increase is reflected in the MCPR operating limit. No other increase in this limit is required as core-wide transients are bounded by the rated power/flow analyses performed for each cycle. Recommendations are provided to adjust the recirculation flow-rate dependent rod block and scram setpoint equations given in the technical specifications for one pump operation. The least stable power/flow condition, the natural circulation point achieved by tripping both recirculation pumps, is not affected by one pump operation.

Under single loop operation, the flow control should be in master manual, since control oscillations may occur in the recirculation flow control system under these conditions.

The derived MAPLHGR reduction factor is 0.83 for the following fuel types: 8x8, 8x8R and P8x8R.

The analyses were performed assuming the equalizer valve was closed. The discharge valve in the idle recirculation loop is normally closed, but if its closure is prevented, the suction valve in the loop should be closed to prevent the loss of Low Pressure Coolant Injection (LPCI) flow out of a postulated break in the idle suction line.

2. MCPR FUEL CLADDING INTEGRITY SAFETY LIMIT

The basis for the MCPR fuel cladding integrity safety limit was derived independent of whether the coolant flow to the core is provided by one or two recirculation pumps. Sufficient margin was included in this basis to account for uncertainties in monitoring the reactor core operating state. The two core operating parameters that might be affected during single loop operation are the total core flow measurements and the TIP readings. The possible effects on these parameters during single loop operation are discussed in this section.

Uncertainties used in the two loop operation analysis are documented in Table S.2-1 of Reference 1 for reloads. A 6% core flow measurement uncertainty has been established for single loop operation (compared to 2.5% for two-loop operation). As shown below, this value conservatively reflects the one standard deviation (one sigma) accuracy of the core flow measurement system documented in Reference 2. The random noise component of the TIP reading uncertainty was revised for single recirculation loop operation to reflect the operating plant test results given in Subsection 2.2 below. This revision resulted in a single loop operation process computer uncertainty of 9.1% for reload cores. A comparable two loop process computer uncertainty value is 8.7% for reload cores. The net effect of the revised core flow and TIP uncertainties is a 0.01 incremental increase in the required MCPR fuel cladding integrity safety limit.

2.1 CORE FLOW UNCERTAINTY

2.1.1 Core Flow Measurement During Single Loop Operation

The jet pump core flow measurement system is calibrated to measure core flow when both sets of jet pumps are in forward flow; total core flow is the sum of the indicated loop flows. For single loop operation, however, the inactive jet pumps may be flowing in reverse (backflowing). Therefore, to obtain the total core flow the measured flow in the backflowing jet pumps must be subtracted from the measured flow in the active loop. In addition, the jet pump flow coefficient is different for reverse flow than for forward flow, and the indicated reverse flow must be modified to account for this difference.

For single loop operation, the total core flow is derived by the following formula:

$$\begin{bmatrix}\text{Total Core} \\ \text{Flow}\end{bmatrix} = \begin{bmatrix}\text{Active Loop} \\ \text{Indicated Flow}\end{bmatrix} - C \begin{bmatrix}\text{Inactive Loop} \\ \text{Indicated Flow}\end{bmatrix}$$

where C (=0.95) is defined as the ratio of "Inactive Loop True Flow" to "Inactive Loop Indicated Flow." "Loop Indicated Flow" is the flow indicated by the jet pump "single-tap" loop flow summers and indicators, which are set to indicate forward flow correctly. The 0.95 factor is the result of a conservative analysis to appropriately modify the single-tap flow coefficient for reverse flow. The expected value of this factor is ~0.88. Therefore, if a more exact (or less conservative) core flow measurement is required, special in-reactor calibration tests would have to be made. Such calibration tests would involve calibration of core support plate ΔP versus core flow during two pump operation along the 100% flow control line and during one pump operation along the 100% flow control line. The correct value of C can then be determined based on the core flow derived from the core support plate ΔP and the loop flow indicator readings.

2.1.2 Core Flow Uncertainty Analysis

The analysis procedure used to establish the core flow uncertainty for one pump operation is essentially the same as for two pump operation, except for some extensions. The core flow uncertainty analysis is described in Reference 2. The analysis of one pump core flow uncertainty is summarized below.

For single loop operation, the total core flow can be expressed as follows (refer to Figure 2-1):

$$W_C = W_A - W_I$$

where

W_C = total core flow;

W_A = active loop flow; and

W_I = inactive loop (true) flow.

By applying the "propagation of errors" method to the above equation, the variance of the total flow uncertainty can be approximated by:

$$\sigma^2_{W_C} = \sigma^2_{W_{sys}} + \left(\frac{1}{1-a}\right)^2 \sigma^2_{W_{A_{rand}}} + \left(\frac{a}{1-a}\right)^2 \left(\sigma^2_{W_{I_{rand}}} + \sigma^2_C\right)$$

where

σ_{WC} = uncertainty of total core flow;

σW_{sys} = uncertainty systematic to both loops;

$\sigma W_{A_{rand}}$ = random uncertainty of active loop only;

$\sigma W_{I_{rand}}$ = random uncertainty of inactive loop only;

σ_C = uncertainty of "C" coefficient; and

a = ratio of inactive loop flow (W_I) to active loop flow (W_A).

Resulting from an uncertainty analysis, the conservative, bounding values of σW_{sys}, $\sigma W_{A_{rand}}$, $\sigma W_{I_{rand}}$ and σ_C are 1.6%, 2.6%, 3.5%, and 2.8%, respectively.

Based on the above uncertainties and a bounding value of 0.38 for "a", the variance of the total flow uncertainty is approximatly:

$$\sigma^2_{W_C} = (1.6)^2 + \left(\frac{1}{1-0.38}\right)^2 (2.6)^2 + \left(\frac{0.38}{1-0.38}\right)^2 \left[(3.5)^2 + (2.8)^2\right] = (5.3\%)^2 .$$

When the effect of 4.1% core bypass flow uncertainty at 12% (bounding case) bypass flow fraction is added to the above total core flow uncertainty, the active coolant flow uncertainty is:

$$\sigma^2_{\substack{\text{active}\\ \text{coolant}}} = (5.3\%)^2 + \left(\frac{0.12}{1-0.12}\right)^2 (4.1\%)^2 \cong (5.3\%)^2$$

which is less than the 6% core flow uncertainty assumed in the statistical analysis.

In summary, core flow during one pump operation is determined in a conservative way, and its uncertainty has been conservatively evaluated.

2.2 TIP READING UNCERTAINTY

To ascertain the TIP noise uncertainty for single recirculation loop operation, a test was performed at an operating BWR. The test was performed at a power level 59.3% of rated with a single recirculation pump in operation (core flow 46.3% of rated). A rotationally symmetric control rod pattern existed prior to the test.

Five consecutive traverses were made with each of five TIP machines, giving a total of 25 traverses. Analysis of this data resulted in a nodal TIP noise of 2.85%. Use of this TIP noise value as a component of the process computer total uncertainty results in a one-sigma process computer total uncertainty value for single loop operation of 9.1% for reload cores.

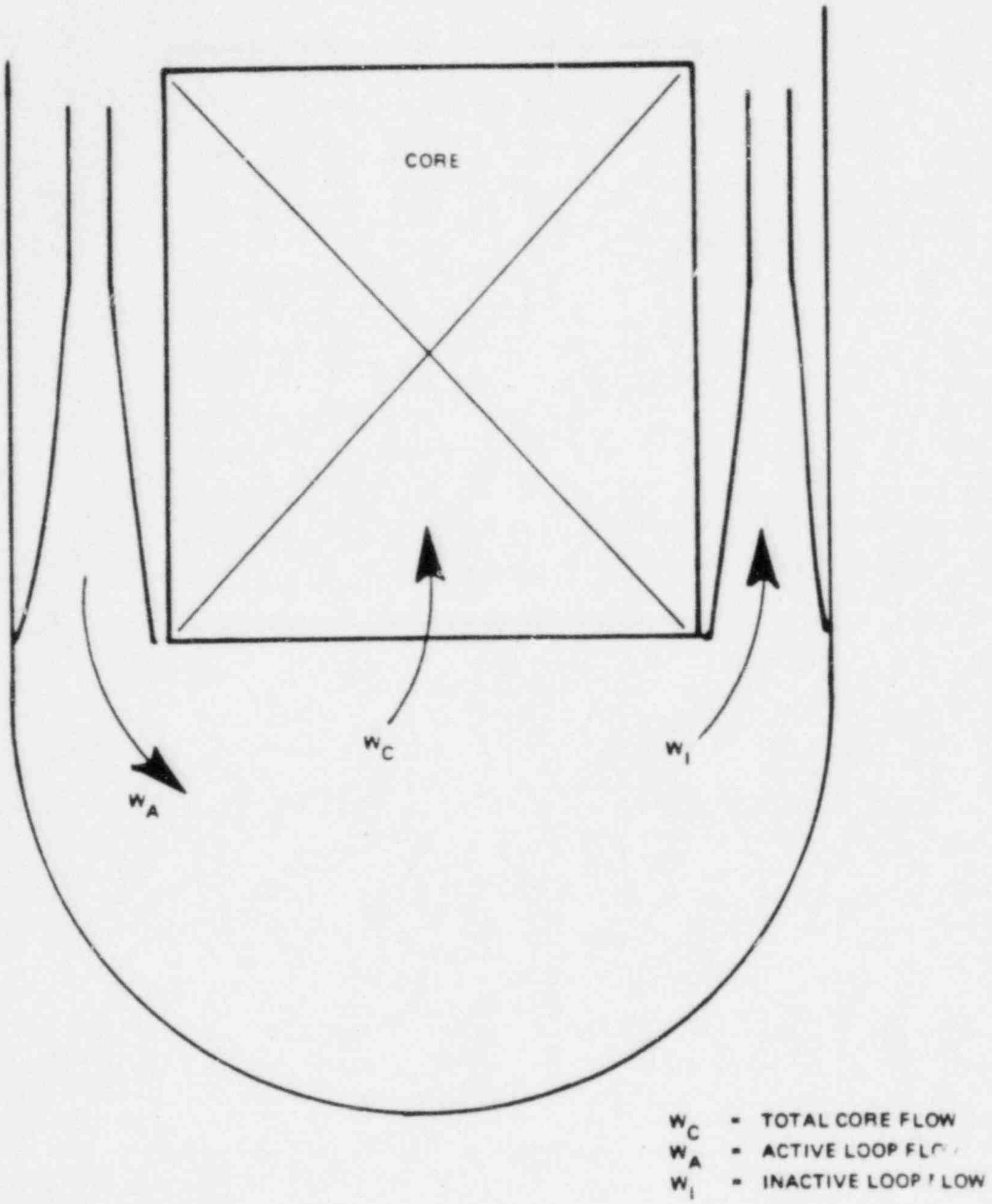

Figure 2-1. Illustration of Single Recirculation Loop Operation Flows

3. MCPR OPERATING LIMIT

3.1 CORE-WIDE TRANSIENTS

Operation with one recirculation loop results in a maximum power output which is approximately 20% to 40% below that which is attainable for two pump operation. Therefore, the consequences of abnormal operational transients from one loop operation will be considerably less severe than those analyzed from a two loop operational mode. For pressurization, flow decrease, and cold water increase transients, previously transmitted Reload and Final Safety Analysis Report (FSAR) results bound both the thermal and overpressure consequences of one loop operation.

Figure 3-1 shows the consequences of a typical pressurization transient (turbine trip) as a function of power level. As can be seen, the consequences of the transient during one loop operation are considerably less severe because of the associated reduction in operating power level.

The consequences of flow decrease transients are also bounded by the full power analysis. A single pump trip from one loop operation is less severe than a two pump trip from full power because of the reduced initial power level.

Cold water increase transients can result from either pump speed-up or introduction of colder water into the reactor vessel by events such as loss of the feedwater heater. For the former, the flow-adjustment factor, K_f is derived by assuming that both recirculation loops increase speed to the maximum permitted by the M-G set scoop tube position. This condition produces the maximum possible power increase and hence maximum ΔCPR for transients initiated from less than rated power and flow. When operating with only one recirculation loop, the flow and power increase associated with the increased speed on only one M-G set will be less than that associated with both pumps increasing speed; therefore, the K_f factors derived with the two pump assumption are conservative for single loop operation. For the latter, the loss of

feedwater heater event is generally the most severe cold water increase event with respect to the increase in core power. This event is caused by positive reactivity insertion due to the increased core flow inlet subcooling; therefore, the event is independent of two pump or one pump operation. The severity of the event is primarily dependent on the initial power level. The higher the initial power level, the greater the CPR change during the transient. Since the initial power level during one pump operation will be significantly lower, the one pump cold water increase case is conservatively bounded by the full power (two pump) analysis.

From the above discussions, it can be concluded that the transient consequences during one loop operation are bounded by those established by previously performed full power analysis. The maximum power level that can be attained in one loop operation is only restricted by the MCPR and overpressure limits established from a full power analysis.

3.2 ROD WITHDRAWAL ERROR

Analysis of the rod withdrawal error (RWE) at rated power is given in the FSAR for the initial core and in cycle-dependent reload supplemental submittals. These analyses are performed to demonstrate that, even if the operator ignores all instrument indications and the alarms which could occur during the course of the transient, the Rod Block Monitor (RBM) system will stop rod withdrawal at a minimum critical power ratio which is higher than the fuel cladding integrity safety limit.

However, one pump operation can result in backflow through 10 of the 20 jet pumps while the flow is being supplied into the lower plenum from the 10 active jet pumps. Because of the potential for backflow through the inactive jet pumps, the direct active-loop flow measurement may not indicate actual flow. Therefore, it is necessary to adjust the flow-biased RBM equation for single loop operation to account for the backflow which is expected in the inactive loop jet pumps, as the active loop jet pump drive flow is increased. This adjustment accounts for the discrepancy between actual flow and indicated

flow in the active loop and preserves the original relationship between rod block and actual effective drive flow when operating with a single recirculation loop.

A procedure was developed for correcting the rod block monitor equation for single loop operation and is described below:

The two pump rod block monitor equation is:

$$RB = mW + [RB_{100} - m(100)]$$

The one pump equation becomes:

$$RB = mW + [RB_{100} - m(100)] - m\Delta W$$

where

ΔW = difference, determined by utility, between two loop and single loop effective drive flow at the same core flow;

RB = power at rod block in %;

m = flow reference slope for the RBM;

W = drive flow in % of rated; and

RB_{100} = top level rod block at 100% flow.

If the rod block monitor setpoint (RB_{100}) is changed, the equation must be recalculated using the new value.

The APRM trip settings are flow biased in the same manner as the rod block monitor trip setting. Therefore, the APRM rod block and scram trip settings are subject to the same procedural changes as the rod block monitor trip setting discussed above.

The combination of the flow-biased corrections described above and the lower reactor power attainable in single loop operation provides assurance that the MCPR safety limit will not be violated during the postulated RWE.

3.3 OPERATING MCPR LIMIT

For single loop operation, the rated condition steady-state MCPR limit is increased by 0.01 to account for the increase in the fuel cladding integrity safety limit (Section 2). At lower flows, the steady-state operating MCPR limit is conservatively established by multiplying the rated flow steady-state limit by the flow adjustment K_f factor. This ensures that the 99.9% statistical limit requirement is always satisfied for any postulated abnormal operational occurrence.

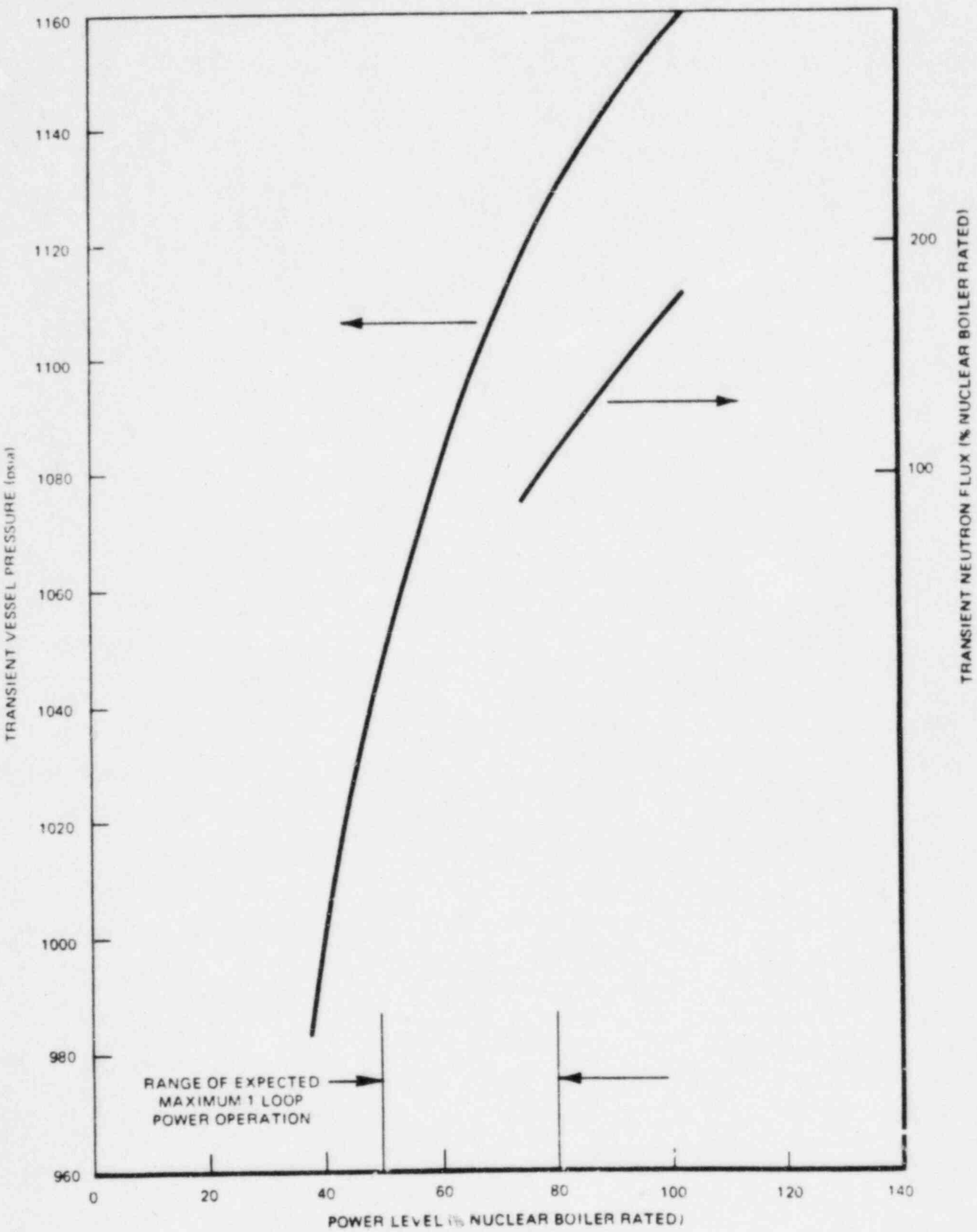

Figure 3-1. Main Turbine Trip with Bypass Manual Flow Control

4. STABILITY ANALYSIS

The least stable power/flow condition attainable under normal conditions occurs at natural circulation with the control rods set for rated power and flow. This condition may be reached following the trip of both recirculation pumps. As shown in Figure 4-1 for a typical BWR, operating along the minimum forced recirculation line with one pump running at minimum speed results in a smaller decay ratio, and is therefore more stable than operation with natural circulation flow only. However, it is slightly less stable than operation with both pumps operating at minimum speed. Under single loop operation, the flow control should be in master manual, since control oscillations may occur in the recirculation flow control system under single loop conditions.

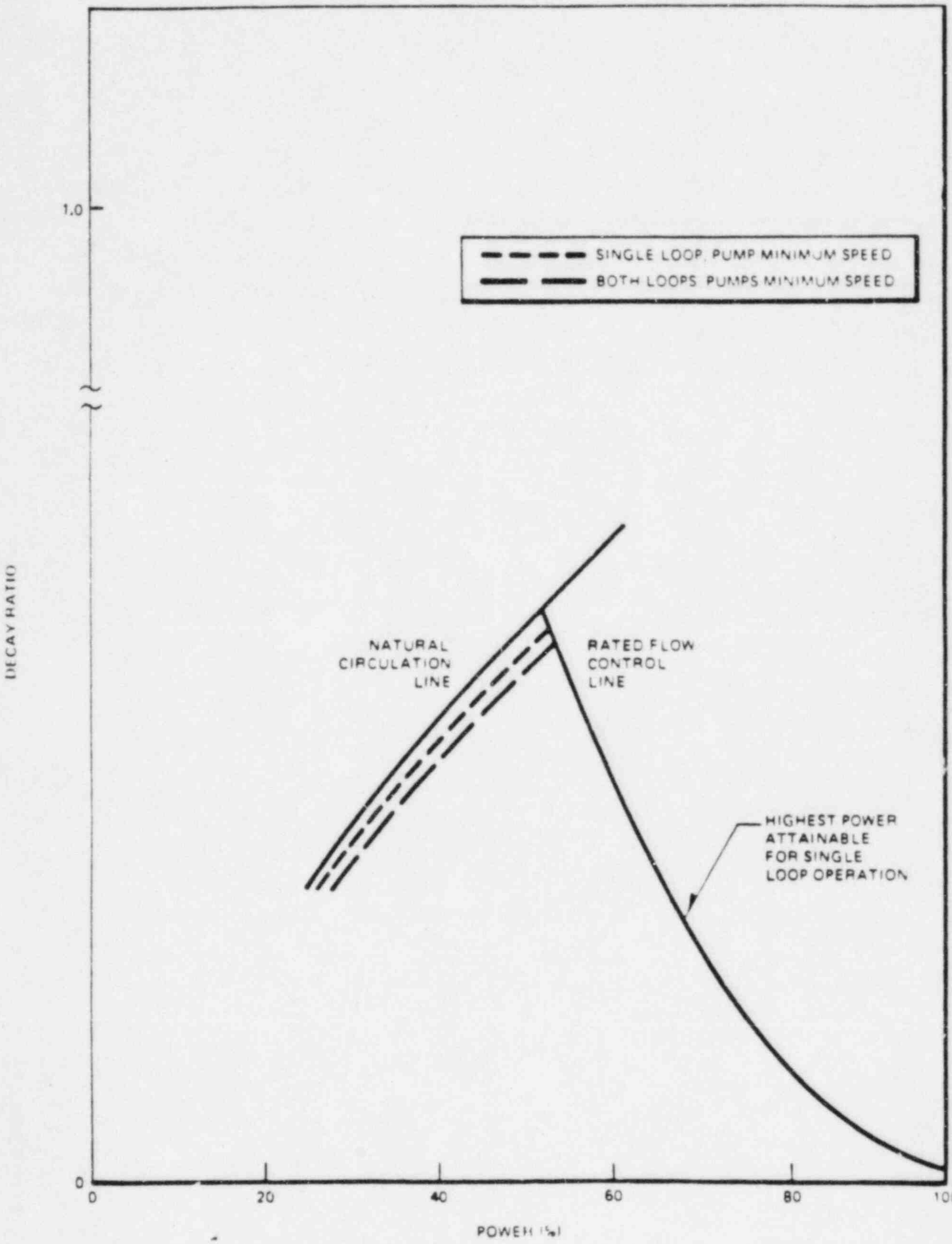

Figure 4-1. Typical Decay Ratio versus Power Curve for Two-Loop and Single Loop Operation

5. ACCIDENT ANALYSES

The broad spectrum of postulated accidents is covered by six categories of design basis events. These events are the loss-of-coolant, recirculation pump seizure, control rod drop, main steam line break, and refueling and fuel assembly loading accidents. The analytical results for loss-of-coolant and recirculation pump seizure accidents with one recirculation pump operating are given below. The results of the two loop analysis for the last four events conservatively bound those for one pump operation.

5.1 LOSS-OF-COOLANT ANALYSIS

A single loop operation analysis utilizing the models and assumptions documented in References 1 and 3 was performed for Vermont Yankee. Using this method, SAFE/REFLOOD code evaluations were made for a full spectrum of break sizes for both the suction and discharge side breaks. The core reflooding minus core uncovery time for the single loop analysis is similar to the two loop analysis. The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) curves currently applied to Vermont Yankee were modified by derived reduction factors for use during one recirculation pump operation.

5.1.1 Break-Spectrum Analysis

A break-spectrum analysis for single loop operation was performed for Vermont Yankee using the model, assumptions and procedure documented in Reference 3. This analysis is necessary because the coastdown flow from the unbroken recirculation loop (which would occur during two loop operation) would not be available during a postulated large break LOCA in the active recirculation loop during single loop operation. This could cause an earlier boiling transition to occur in the core which would increase the calculated Peak Cladding Temperature (PCT). Thus, single loop LOCA calculations are performed assuming a bounding (early) boiling transition time (Reference 3) which leads to revised MAPLHGR limits.

The suction break spectrum core uncovery and reflooding times for single loop operation are compared to the previously performed two loop operation in Figures 5-1 and 5-2, respectively. The total core uncovered time (reflooding time minus uncovery time) for the suction break is compared in Figure 5-3.

The most limiting break size for both two loop and single loop operation is the 100% DBA break of the discharge line. The break spectrums depicting both single loop and two loop core uncovery and reflooding times corresponding to the discharge line break are shown in Figures 5-4 and 5-5, respectively. The total core uncovered time is shown in Figure 5-6 for the discharge break spectrums.

5.1.2 Single Loop MAPLHGR Determination

The small differences in uncovered time and reflooding time for the limiting break size would result in a small difference in the calculated peak cladding temperature. Therefore, as noted in Reference 3, the one and two loop SAFE/REFLOOD results can be considered similar and the generic alternative procedure described in Section II.A.7.4 of this reference was used to calculate the MAPLHGR reduction factors for single loop operation.

MAPLHGR reduction factors were determined for the 100% DBA discharge break. The most limiting reduction factors for each fuel type are shown in Table 5-1. One loop operation MAPLHGR values are derived by multiplying the current two loop operation MAPLHGR values by the reduction factor for that fuel type. As discussed in Reference 3, single recirculation loop MAPLHGR values are conservative when calculated in this manner.

5.1.3 Small-Break Peak Cladding Temperature

Section II.A.7.4.4.2 of Reference 3 discusses the small sensitivity of the calculated Peak Cladding temperature (PCT) to the duration of nucleate boiling and to the assumptions used in the one pump operation analysis. While there is a potential for a slight increase (~50°F) in PCT, this increase

is more than offset by the decreased MAPLHGR (equivalent to 300° to 350°F PCT) for one pump operation. Therefore, the calculated PCT values for small breaks are significantly below the 2200°F cladding temperature limit specified in 10CFR50.46 and are not limiting.

5.2 ONE PUMP SEIZURE ACCIDENT

The pump seizure event is a very mild accident in relation to other accidents such as the Loss-of-Coolant Accident (LOCA). This has been demonstrated by analyses in Reference 2 for the case of two pump operation, and that it is also true for the case of one pump operation is easily verified by consideration of the two events. In both accidents, the recirculation driving loop flow is lost rapidly. In the case of the pump seizure, stoppage of the pump occurs; for the LOCA, the severance of the line has a similar, but more rapid and severe influence. Following a pump seizure event, natural circulation flow continues, water level is maintained, the core remains submerged, and the combined effects provide a continuous core cooling mechanism. However, for the LOCA, complete flow stoppage occurs and the water level decreases as a result of loss-of-coolant, resulting in uncovery of the reactor core and subsequent heatup of the fuel rod cladding. In addition, for the pump seizure accident, reactor pressure does not decrease, whereas complete depressurization occurs for the LOCA. Clearly, the increased temperature of the cladding and reduced reactor pressure for the LOCA both combine to yield a much more severe stress and potential for cladding perforation for the LOCA than for the pump seizure. Therefore, it can be concluded that the potential effects of the hypothetical pump seizure accident are very conservatively bounded by the effects of a LOCA.

Table 5-1

LIMITING MAPLHGR REDUCTION FACTORS

Fuel Type	Reduction Factor
8x8	0.83
8x8R	0.83
P8x8R	0.83

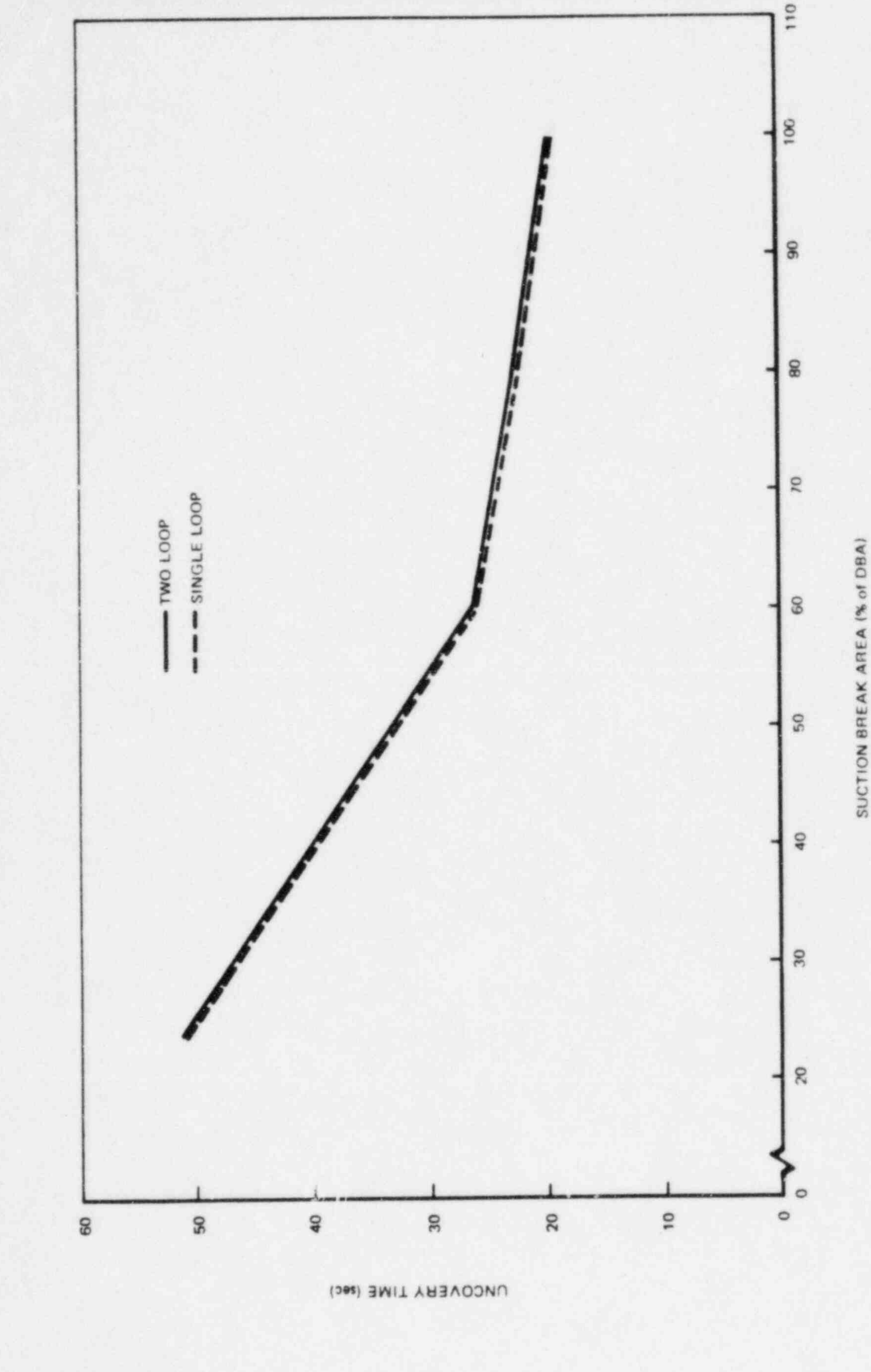

Figure 5-1. Vermont Yankee Core Uncovery Time versus Suction Break Area

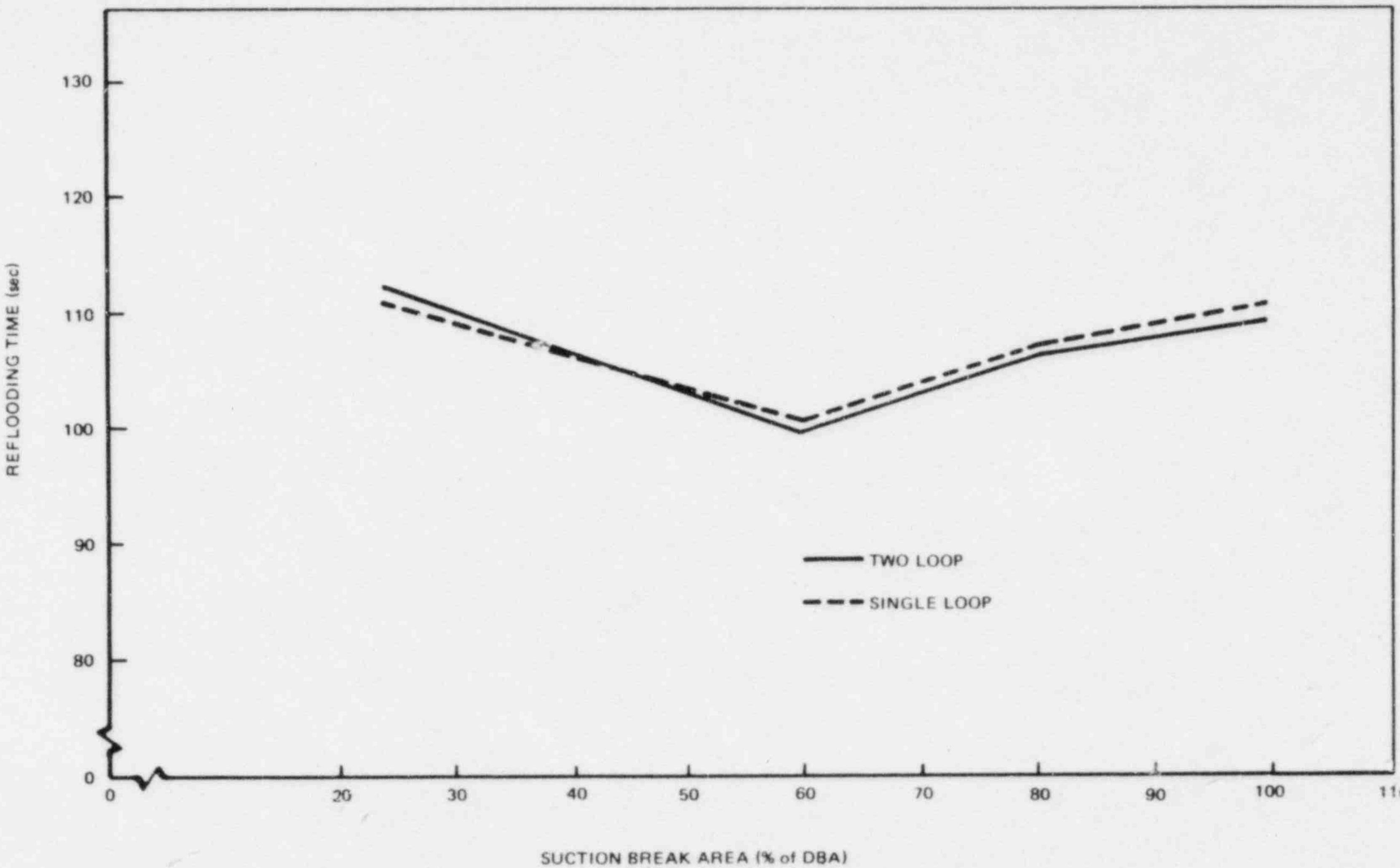

Figure 5-2. Vermont Yankee Core Reflooding Time versus Suction Break Area

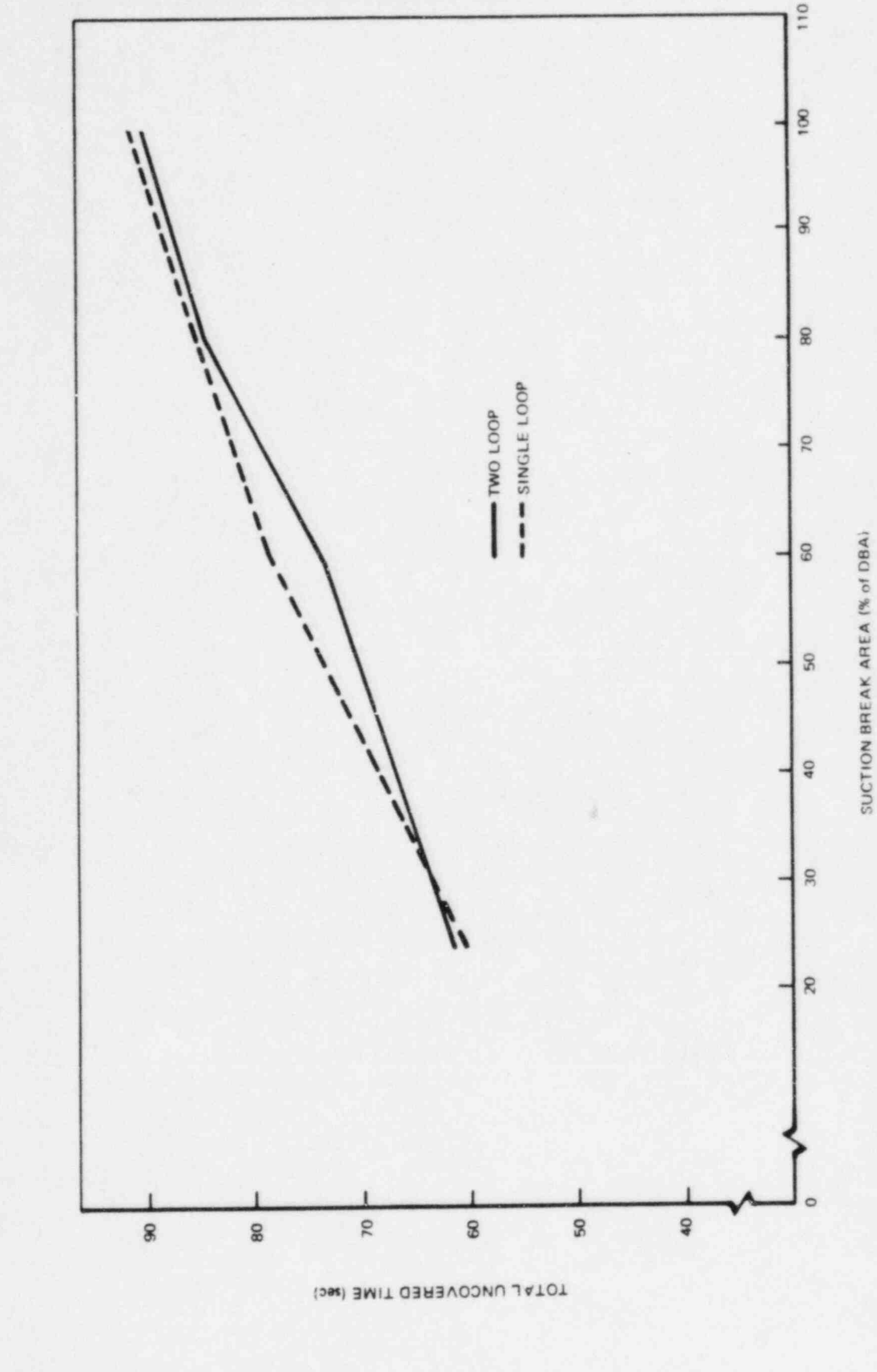

Figure 5-3. Vermont Yankee Core Total Uncovered Time versus Suction Break Area

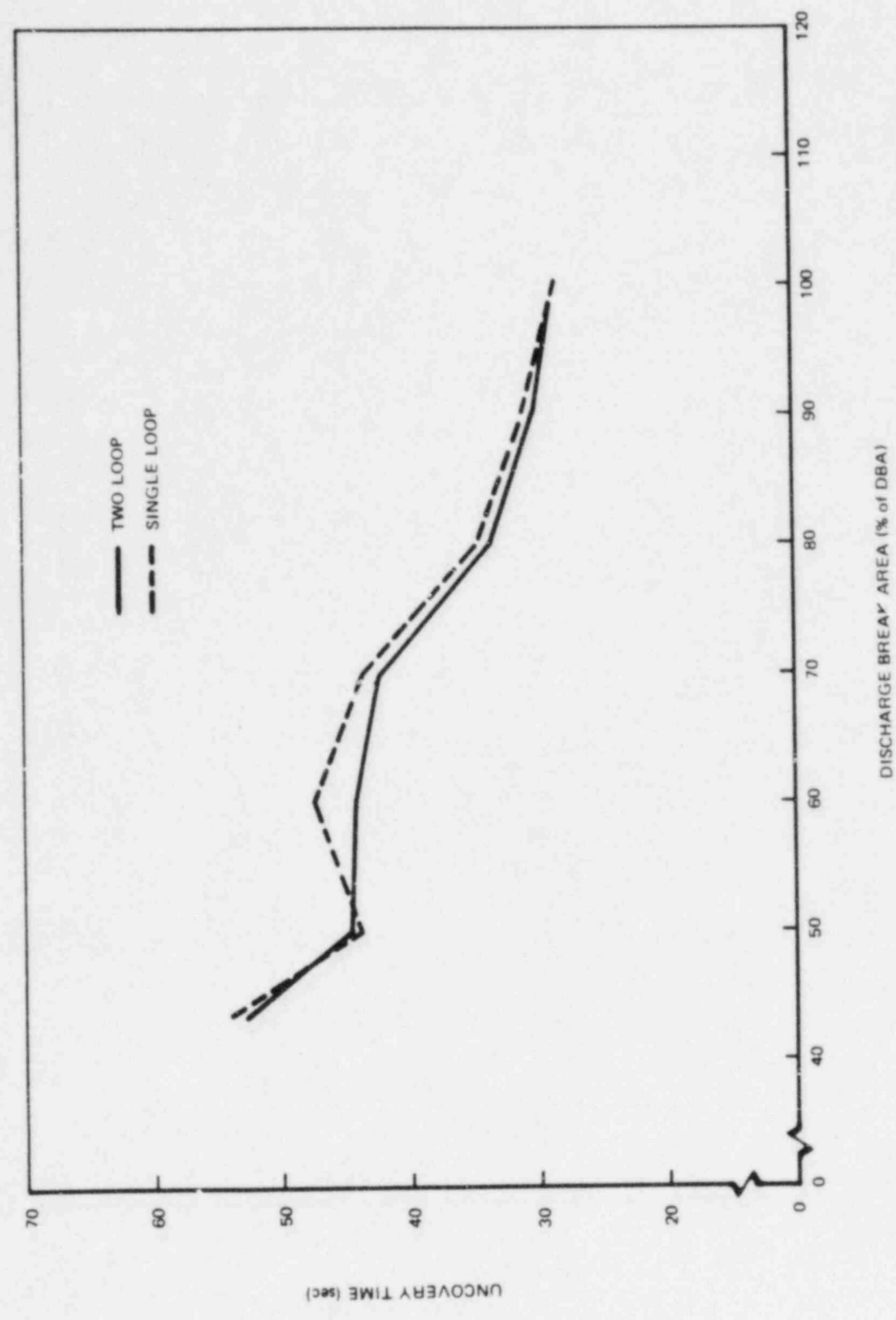

Figure 5-4. Vermont Yankee Core Uncovery Time versus Discharge Break Area

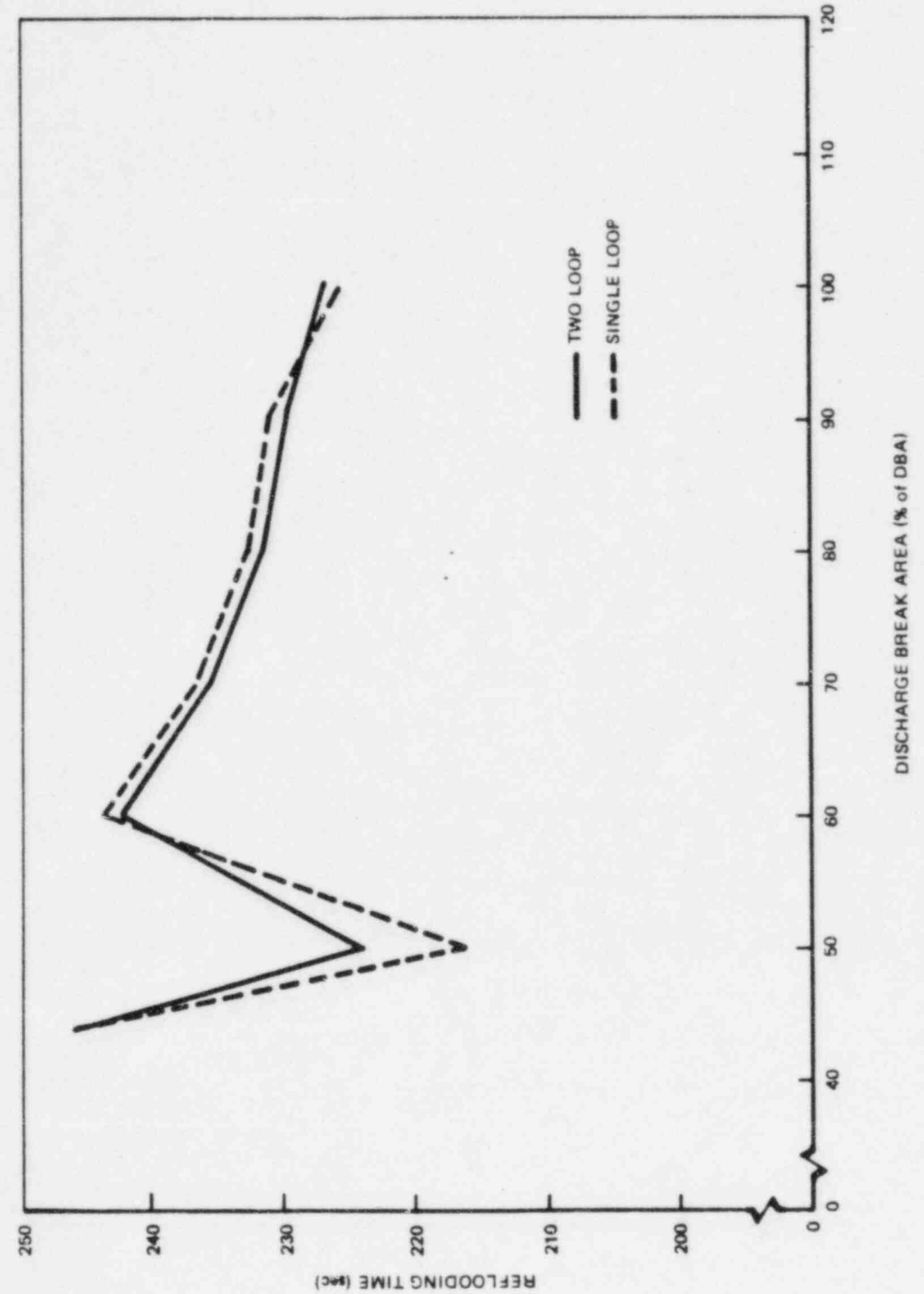

Figure 5-5. Vermont Yankee Core Reflooding Time Versus Discharge Break Area

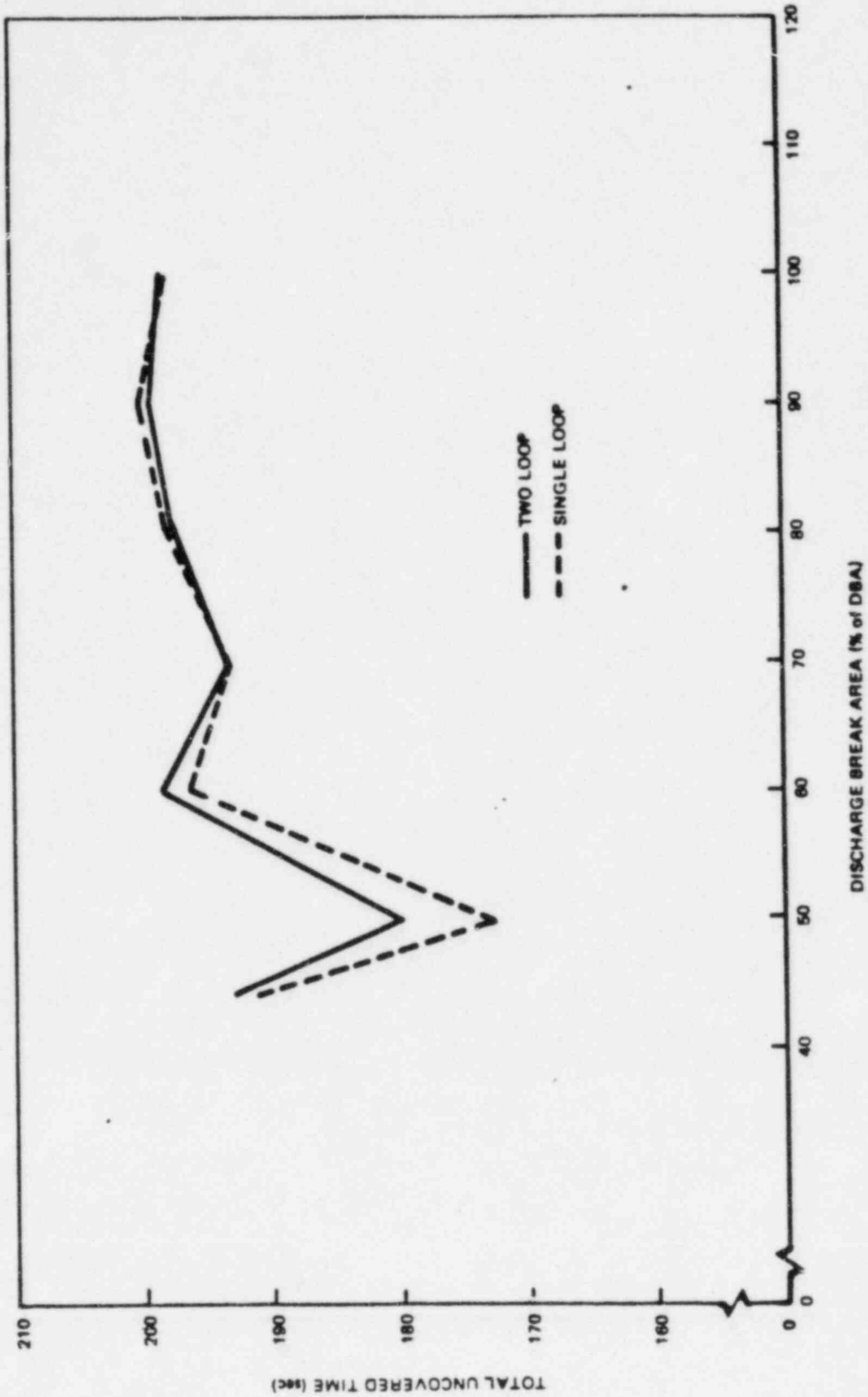

Figure 5-6. Vermont Yankee Core Total Uncovered Time versus Discharge Break Area

6. REFERENCES

1. "General Electric Standard Application for Reactor Fuel," General Electric Company, January 1982 (NEDE-24011-P-A-4), and "United States Supplement," General Electric Company, January 1982 (NEDE-24011-P-A-4-US).

2. "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application," General Electric Company, January 1977 (NEDO-10958-A).

3. "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K Amendment No. 2 - One Recirculation Loop Out-of-Service," General Electric Company, Revision 1, July 1978 (NEDO-20566-2).