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MONTICELLO NUCLEAR GENERATING PLANT SINGLE-LOOP OPERATION

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ITEM	REFERENCES (SECTION, PAGE PARAGRAPH, LINE)	INSTRUCTIONS (CORRECTIONS AND ADDITIONS)
01	Page 1-1/1-2	Replace with new page 1-1/1-2.
02	Page 5-5	Replace with new page 5-5.

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SINGLE-LOOP OPERATION

NUCLEAR ENERGY ENGINEERING DIVISION • GENERAL ELECTRIC COMPANY
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1. INTRODUCTION AND SUMMARY

The current technical specifications for the Monticello Nuclear Generating Plant do not allow plant operation beyond a relatively short period of time if an idle recirculation loop cannot be returned to service. The Monticello Nuclear Generating Plant (Technical Specification 3.6 G) shall not be operated for a period in excess of 24 hours with one recirculation loop out of service.

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 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 Evaluation Reports and Reference 1 were reviewed for one-pump operation. Increased uncertainties in the core total flow and TIP readings resulted in an 0.01 incremental increase in the 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, and the recirculation flow-rate dependent rod block and scram setpoint equations given in the technical specifications are adjusted for one-pump operation. The least stable power/flow condition, achieved by tripping both recirculation pumps, is not affected by one-pump operation.

During single-loop operation the flow control should be in master manual since control oscillations might occur in the recirculation flow control system under automatic flow control conditions.

Derived MAPLHGR reduction factors are 0.85, 0.85, and 0.85 for the 8x8, 8x8R and P8x8R fuel types, respectively.

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 partial loss of Low Pressure Coolant Injection (LPCI) flow through the recirculation pump into the downcomer degrading the intended LPCI performance.

2. MCPR FUEL CLADDING INTEGRITY SAFETY LIMIT

Except for core total flow and TIP reading, the uncertainties used in the statistical analysis to determine the MCPR fuel cladding integrity safety limit are not independent on whether coolant flow is provided by one or two recirculation pumps. Uncertainties used in the two-loop operation analysis are documented in the FSAR for initial cores and in Table 5-1 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. The comparable two-loop process computer uncertainty value is 8.7% for reload cores. The net effect of these two revised 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 will be backflowing. Therefore, 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 measurement of reverse flow must be modified to account for this difference.

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

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

where C (= 0.95) is defined as the ratio of "Inactive Loop True Flow" to "Inactive Loop Indicated Flow," and "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 was the result of a conservative analysis to appropriately modify the single-tap flow coefficient for reverse flow.* If a more exact, less conservative core flow is required, special in-reactor calibration tests would have to be made. Such calibration tests would involve calibrating core support plate ΔP versus core flow during two-pump operation along the 100% flow control line, operating on one pump along the 100% flow control line, and calculating the correct value of C 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 uncertainty 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 (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.

*The expected value of the "C" coefficient is ∿0.88.

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_{W_C} = uncertainty of total core flow;

$\sigma_{W_{sys}}$ = 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).

Resulted from an uncertainty analysis, the conservative, bounding values of $\sigma_{W_{sys}}$, $\sigma_{W_{A_{rand}}}$, $\sigma_{W_{I_{rand}}}$ and σ_C are 1.6%, 2.6%, 3.5% and 2.8%, respectively. Based on above uncertainties and a bounding value of 0.36 for "a", the variance of the total flow uncertainty is approximately:

$$\sigma^2_{W_C} = (1.6)^2 + \left(\frac{1}{1-0.36}\right)^2 (2.6)^2 + \left(\frac{0.36}{1-0.36}\right)^2 \left[(3.5)^2 + (2.8)^2\right]$$

$$= (5.0\%)^2$$

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

$$\sigma^2_{\substack{\text{active}\\\text{coolant}}} = (5.0\%)^2 + \left(\frac{0.12}{1-0.12}\right)^2 (4.1\%)^2 = (5.0\%)^2$$

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

In summary, core flow during one-pump operation is measured 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 their 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.

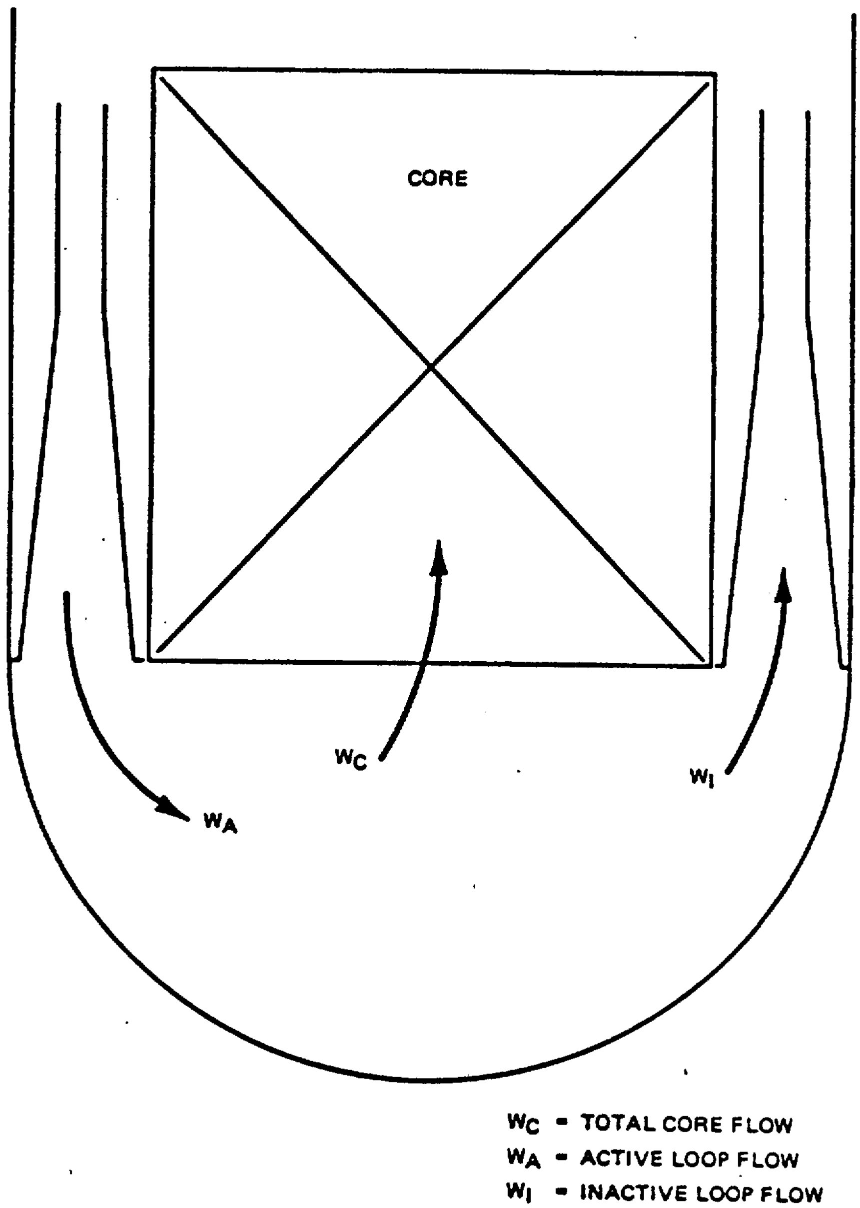

Figure 2-1. Illustration of Single Recirculation Loop Operation Flows

3. MCPR OPERATING LIMIT

3.1 CORE-WIDE TRANSIENTS

Operation with one recirculation loop results in a maximum power output which is 20 to 30% 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/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 one-loop operation are considerably less because of the associated reduction in operating power level.

The consequences from 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 recirculation pump speedup or restart, or introduction of colder water into the reactor vessel by events such as loss of feedwater heater. The K_f factors are derived 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. Inadvertent restart of the idle recirculation pump would result in a neutron flux transient which would exceed the flow reference scram. The resulting scram is expected to be less severe than the rated power/flow case documented in the FSAR. The latter event (loss of feedwater heating) is generally the most severe cold water increase event with respect to increase in core power. This event is caused by positive reactivity insertion from core flow inlet subcooling; therefore, 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 consequence from one-loop operation is bounded by previously submitted full power analysis.

3.2 ROD WITHDRAWAL ERROR

The rod withdrawal error 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 alarm which could occur during the course of the transient, the rod block system will stop rod withdrawal at a minimum critical power ratio (MCPR) which is higher than the fuel cladding integrity safety limit. Correction of the rod block equation (below) and lower power assures that the MCPR safety limit is not violated.

One-pump operation results 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 backflow through the inactive jet pumps, the present rod block equation was conservatively modified for use during one-pump operation because the direct active-loop flow measurement may not indicate actual flow above about 35% drive flow without correction.

A procedure has been established for correcting the rod block equation to account for the discrepancy between actual flow and indicated flow in the active loop. This preserves the original relationship between rod block and actual effective drive flow when operating with a single loop.

The two-pump rod block equation is:

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

The one-pump equation becomes:

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

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 rod block monitor (RBM);

W = drive flow in % of rated; and

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

If the rod block 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.

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 MCPR operating limit is conservatively established by multiplying the rated flow steady-state limit by the K_f factor. This ensures that the 99.9% statistical limit requirement is always satisfied for any postulated abnormal operational transient.

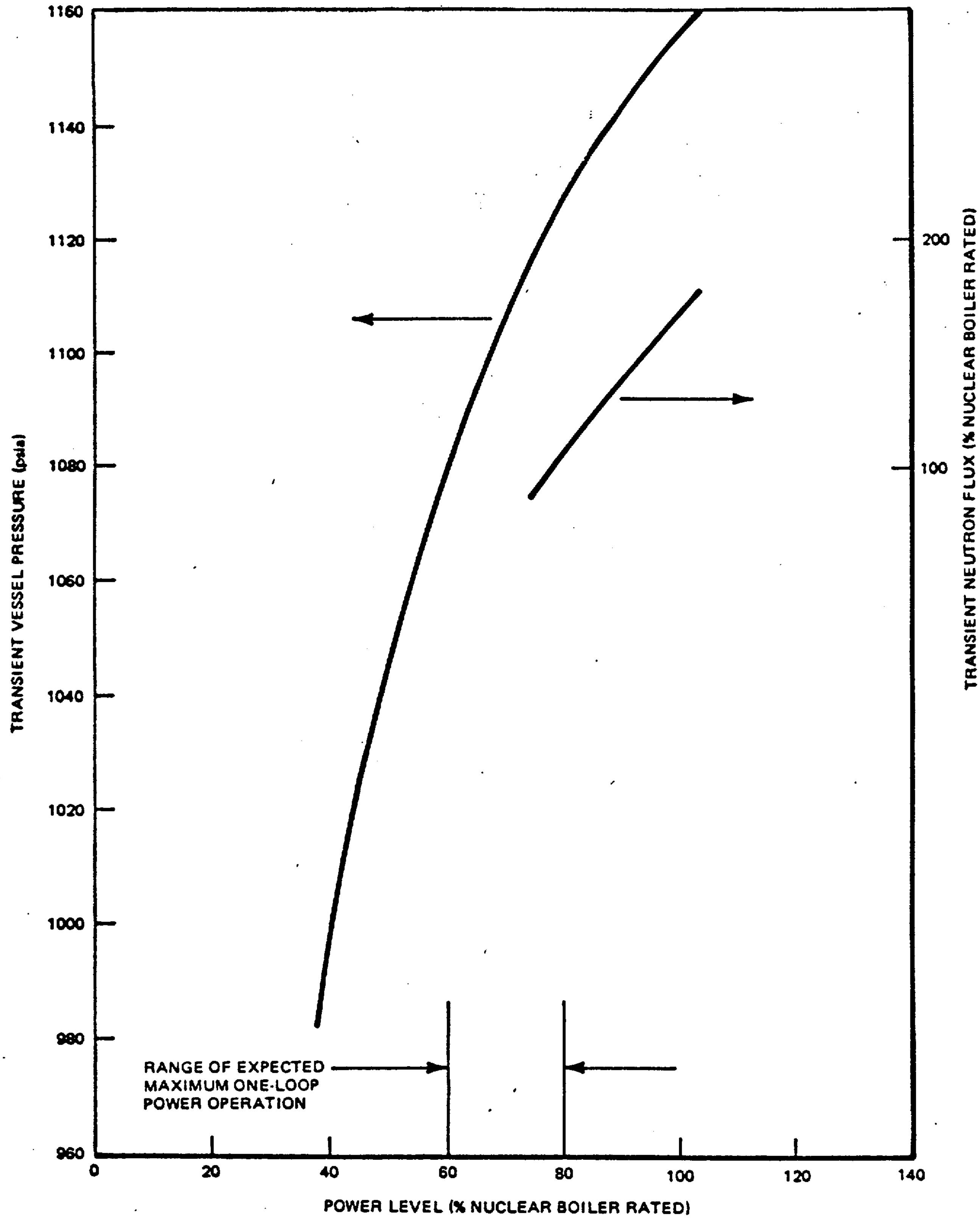

Figure 3-1. Main Turbine Trip with Bypass Manual Flow Control

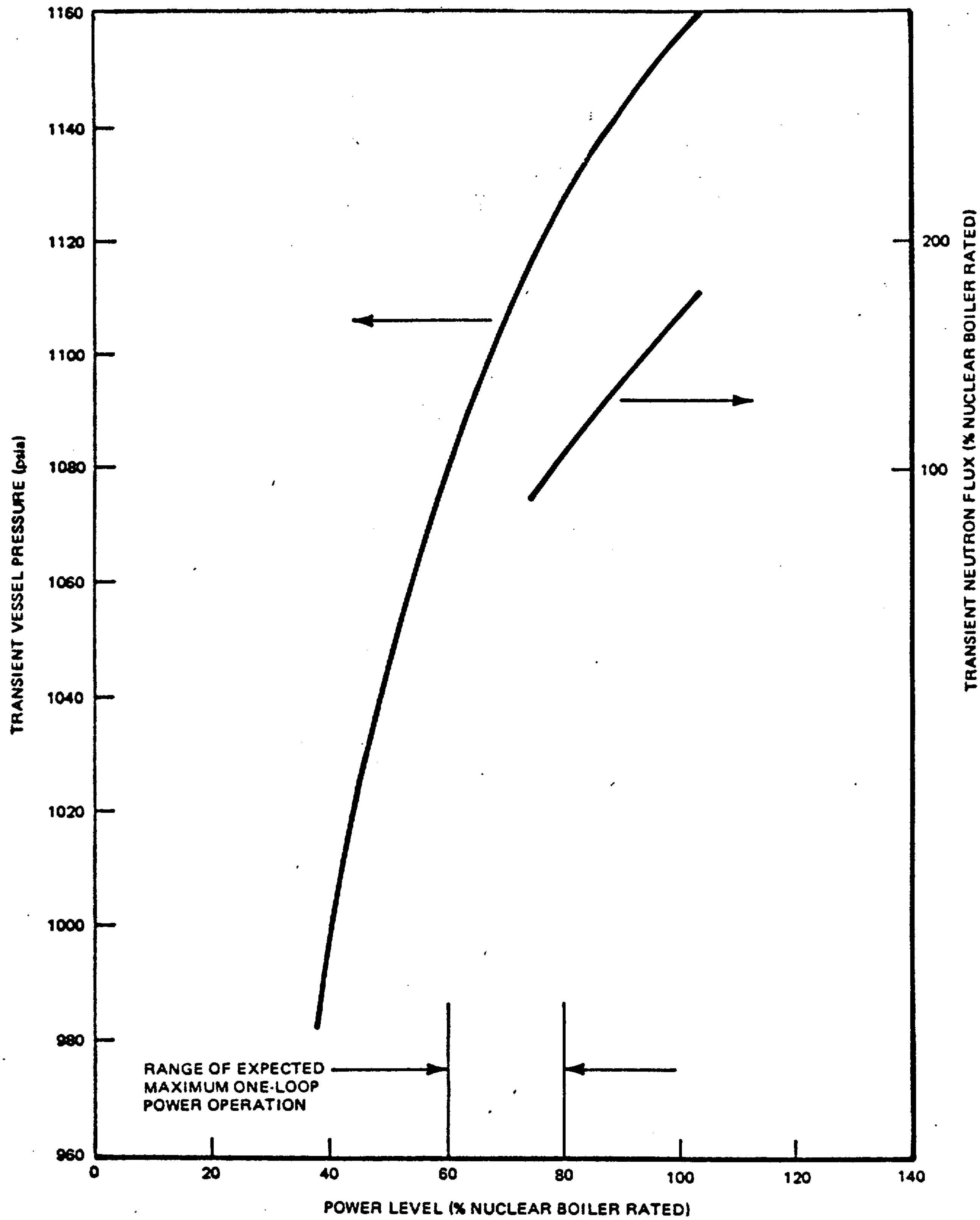

Figure 3-1. Main Turbine Trip with Bypass Manual Flow Control

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, operation along the minimum forced recirculation line with one pump running at minimum speed is more stable than operating with natural circulation flow only, but is less stable than operating with both pumps operating at minimum speed.

During single-loop operation, the flow control should be in master manual, since control oscillations might occur in the recirculation flow control system under automatic flow control conditions.

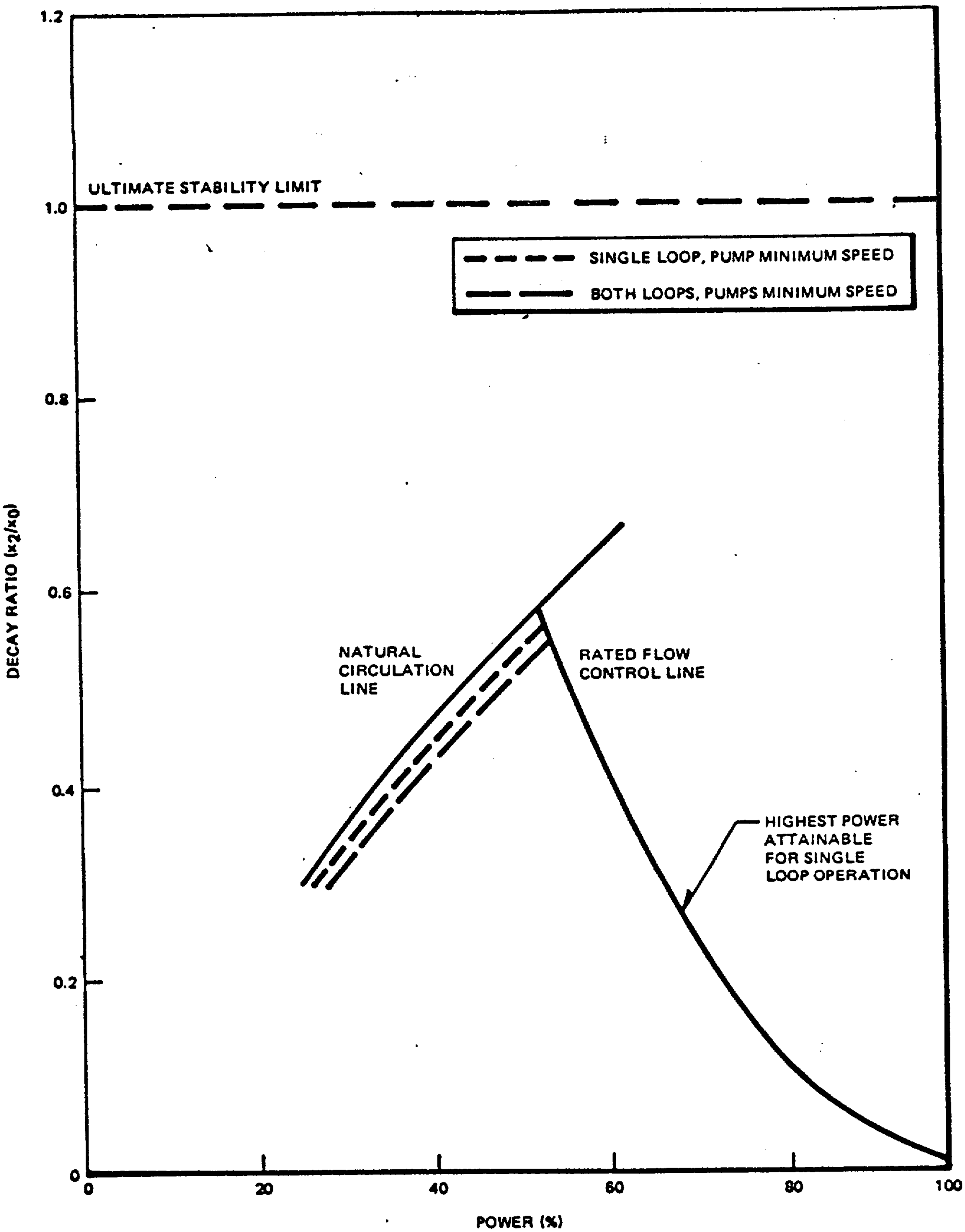

Figure 4-1. Decay Ratio Versus Power Curve for Two-Loop and Single-Loop Operation

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 steamline break, refueling, and fuel assembly loading accidents. The analytical results for the 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 are conservatively applicable for one-pump operation.

5.1 LOSS-OF-COOLANT ACCIDENT ANALYSIS

A single-loop operation analysis utilizing the models and assumptions documented in Reference 3 was performed for the Monticello Nuclear Generating Plant. Using this method, SAFE/REFLOOD computer code runs were made for a full spectrum of break sizes for the suction breaks. Because the reflood 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 were modified by derived reduction factors for use during one recirculation pump operation.

5.1.1 Break Spectrum Analysis

A break spectrum analysis was performed using the SAFE/REFLOOD computer codes and the assumptions given in Section II.A.7.2.2. of Reference 3.

Since the suction break is the most limiting, the suction break spectrum reflood times for one recirculation loop operation are compared to the standard previously performed two-loop operation in Figure 5-1. The uncovered time (reflood time minus recovery time) for the suction break spectrum is compared in Figure 5-2.

For the Monticello Nuclear Generating Plant, the maximum reflooding time for the standard two-loop analysis is 345 seconds with a boiling transition time within 9 sec, occurring at 40% of the Design Basis Accident (DBA) suction break, which is the most limiting break for the two-loop operation. For the single-loop analysis, the maximum reflooding time is 351 seconds, occurring at 40% DBA suction break. These uncovered times can be considered similar.

5.1.2 Single-Loop MAPLHGR Determination

The small differences in uncovered time and reflood time for the limiting break size would result in a small increase 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 cases given in Table 5-1. The most limiting reduction factors for each fuel type is shown in Table 5-2.

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 clad temperature (PCT) to the assumptions used in the one-pump operation analysis and the duration of nucleate boiling. Since the slight increase ($\sim$50°F) in PCT is overwhelmingly offset by the decreased MAPLHGR (equivalent to 300° to 500°F $\sim$PCT) for one pump operation, the calculated PCT values for small breaks will be well below the 2200°F 10CFR50.46 cladding temperature limit.

5.2 ONE-PUMP SEIZURE ACCIDENT

The one-pump seizure accident is a relatively mild event during two recirculation pump operation, as documented in References 1 and 2. Similar analyses were performed to determine the impact this accident would have on one recirculation pump operation. These analyses were performed with the models documented in Reference 1 for a large core BWR/4 plant (Reference 4). The analyses were initialized from steady-state operation at the following initial conditions, with the added condition of one inactive recirculation loop. Two sets of initial conditions were assumed:

(1) Thermal Power = 75% and core flow = 58%

(2) Thermal Power = 82% and core flow = 56%

These conditions were chosen because they represent reasonable upper limits of single-loop operation within existing MAPLHGR and MCPR limits at the same maximum pump speed. Pump seizure was simulated by setting the single operating pump speed to zero instantaneously.

The anticipated sequence of events following a recirculation pump seizure which occurs during plant operation with the alternate recirculation loop out of service is as follows:

(1) The recirculation loop flow in the loop in which the pump seizure occurs drops instantaneously to zero.

(2) Core voids increase which results in a negative reactivity insertion and a sharp decrease in neutron flux.

(3) Heat flux drops more slowly because of the fuel time constant.

(4) Neutron flux, heat flux, reactor water level, steam flow, and feedwater flow all exhibit transient behaviors. However, it is not anticipated that the increase in water level will cause a turbine trip and result in scram.

It is expected that the transient will terminate at a condition of natural circulation and reactor operation will continue. There will also be a small decrease in system pressure.

The minimum CPR for the pump seizure accident for the large core BWR/4 plant was determined to be greater than the fuel cladding integrity safety limit; therefore, no fuel failures were postulated to occur as a result of this analyzed event.

These results are applicable to the Monticello Nuclear Generating Plant.

Table 5-1

MAPLHGR MULTIPLIER CASES

Fuel Type	Cases Calculated
8x8	100% DBA Suction Break
	40% DBA Suction Break*
8x8R/P8x8R	100% DBA Suction Break
	40% DBA Suction Break*

*Most limiting break for MAPLHGR reduction factors.

Table 5-2

LIMITING MAPLHGR REDUCTION FACTORS

Fuel Type	Reduction Factors
8x8	0.85
8x8R	0.85
P8x8R	0.85

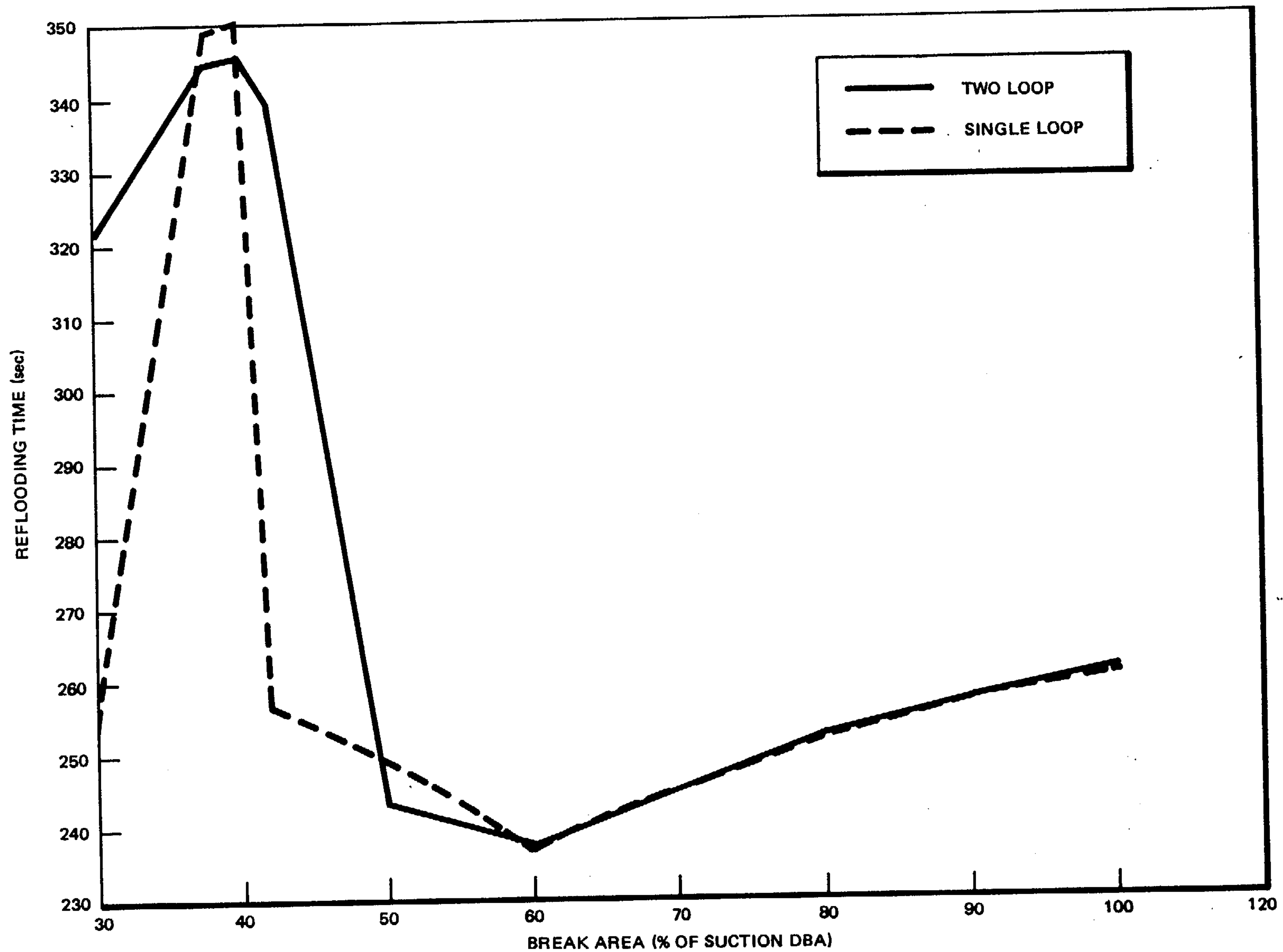

Figure 5-1. Monticello Reflooding Time vs. Break Area

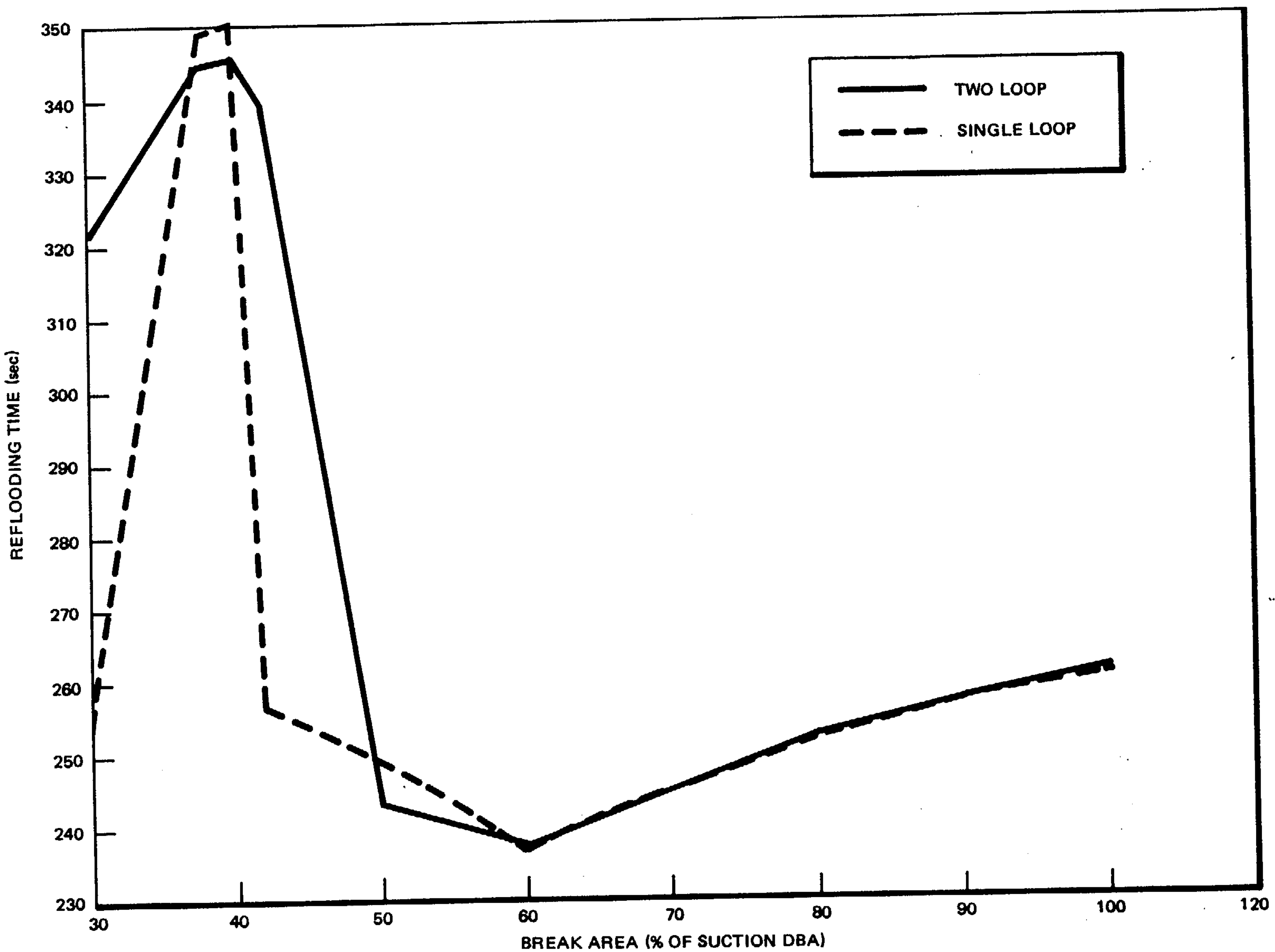

Figure 5-1. Monticello Reflooding Time vs. Break Area

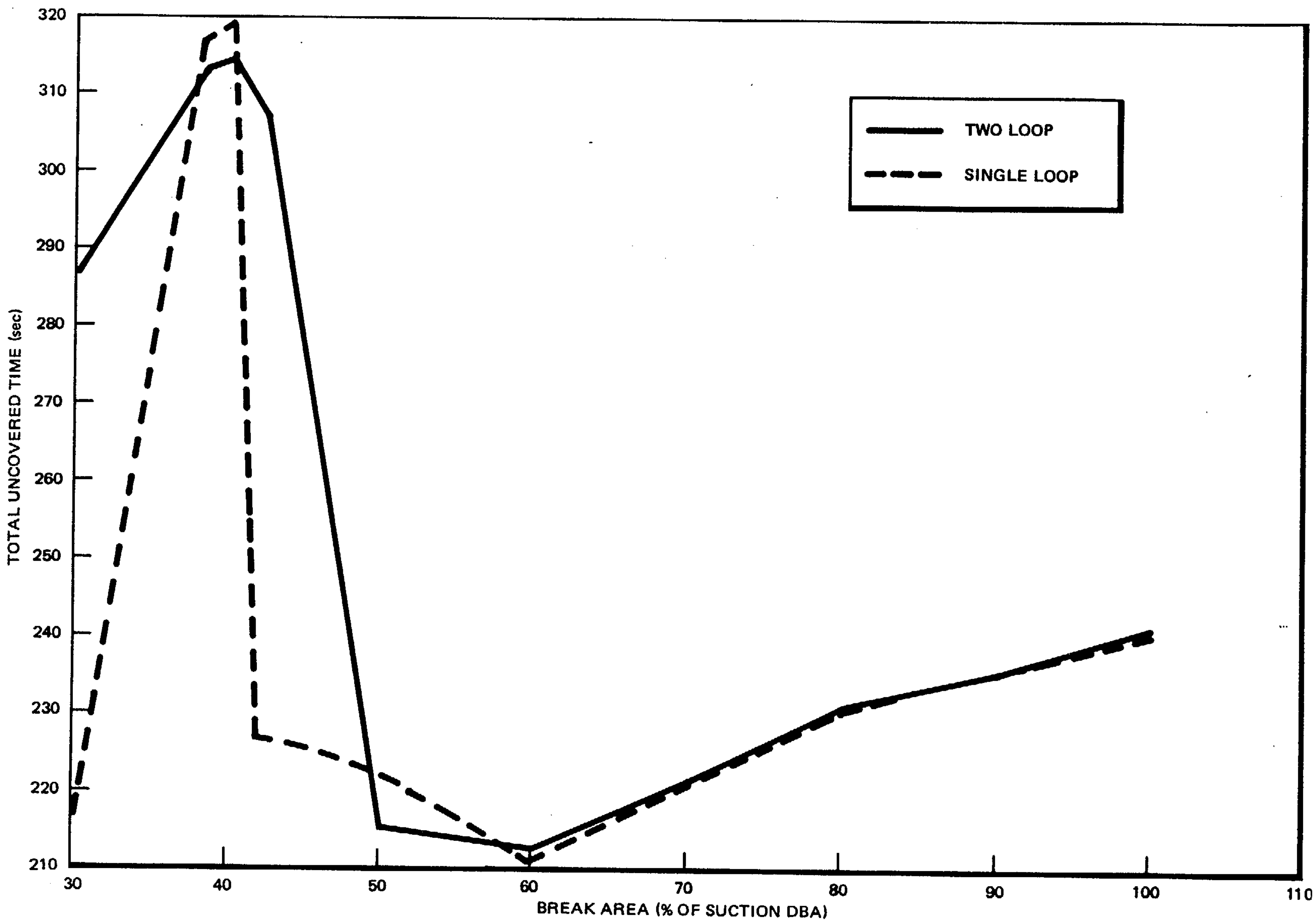

Figure 5-2. Monticello Total Uncovered Time vs. Break Area

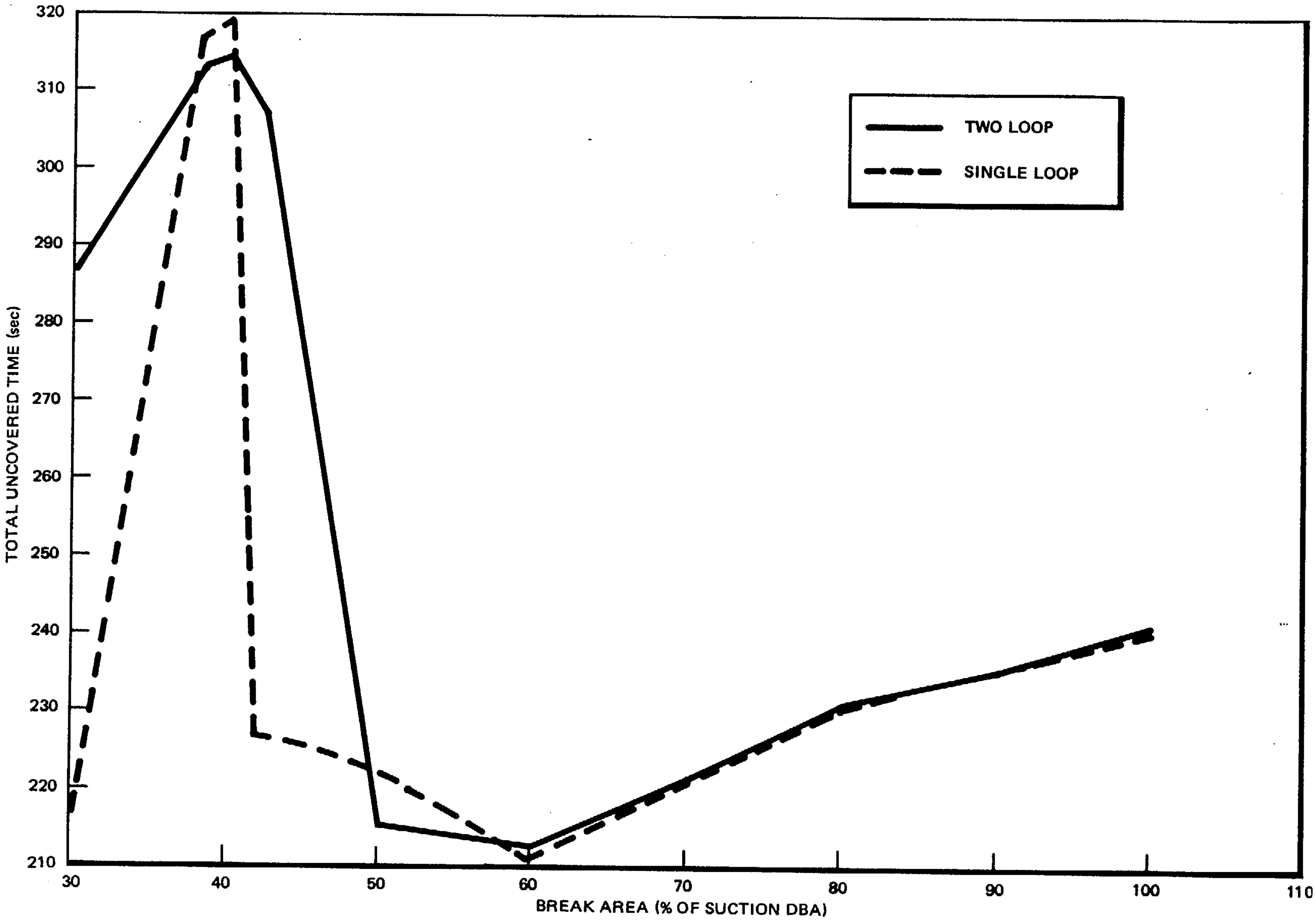

Figure 5-2. Monticello Total Uncovered Time vs. Break Area

6. REFERENCES

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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).

4. Enclosure to Letter #TVA-BFNP-TS-117, O. E. Gray III to Harold R. Denton, September 15, 1978.