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BRUNSWICK STEAM ELECTRIC PLANT UNITS 1 AND 2 SINGLE-LOOP OPERATION

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NUCLEAR POWER SYSTEMS DIVISION • GENERAL ELECTRIC COMPANY
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GENERAL ELECTRIC

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ILLUSTRATIONS

1. INTRODUCTION AND SUMMARY

The current technical specifications for the Brunswick Steam Electric Plant, Units 1 and 2, do not allow plant operation beyond a relatively short period of time if an idle recirculation loop cannot be returned to service. The Brunswick 1 and 2 nuclear power plant (Technical Specification 3.4.1.1) 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 Analysis Reports and Reference 1 were reviewed for one-pump operation. Increased uncertainties in the total core flow and Traversing In-Core Probe (TIP) readings resulted 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 also added to 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 set-point 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. Under single-loop operation, the flow control must be in master manual, since control oscillations may occur in the recirculation flow control system under these conditions. Derived MAPLHGR reduction factors for single recirculation pump operation are given in Table 5-2.

The analyses were performed assuming the two recirculation manifolds are isolated from one another by closure of appropriate valves in the cross-tie (equalizer) line between the loops. 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) through the recirculation pump into the downcomer degrading the intended LPCI performance.

2. MCPR FUEL CLADDING INTEGRITY SAFETY LIMIT

Most of the uncertainties used in the statistical analysis presented in Table 4-2 of Reference 2 are independent of whether flow is provided by two loop or single loop except the core total flow and the TIP reading uncertainties. The one standard deviation (1-sigma) of the core total flow for single loop operation may increase to about 6% of the rated core flow from 2.5% as that for the two-loop operation. The process computer effective TIP reading uncertainty, which considers the single-loop operation TIP data noise measured at Browns Ferry 1, is increased to 6.8% from 6.3% for the initial cycle and to 9.1% from 8.7% for the reload cores. The net effect of these two revised uncertainties is a 0.01 incremental increase in the required General Electric Thermal Analysis Basis (GETAB) safety limit MCPR.

The steady-state operating MCPR limit with single-loop operation is conservatively established by the multiplication of the existing K_f factor and the rated flow steady-state operating limit (which should be increased by 0.01 to reflect the safety limit MCPR change). This ensures that the 99.9% statistical limit requirement is always satisfied.

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 loop 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:

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

where C (= 0.95) is defined as the ratio of "Inactive Loop True Flow" to "Inactive Loop Indicated Flow," 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 measurement 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 (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:

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*The expected value of the "C" coefficient is $\sim$0.88.

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

where

σ_{WC} = uncertainty of total core flow;

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

Resulting 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 the 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 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 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.

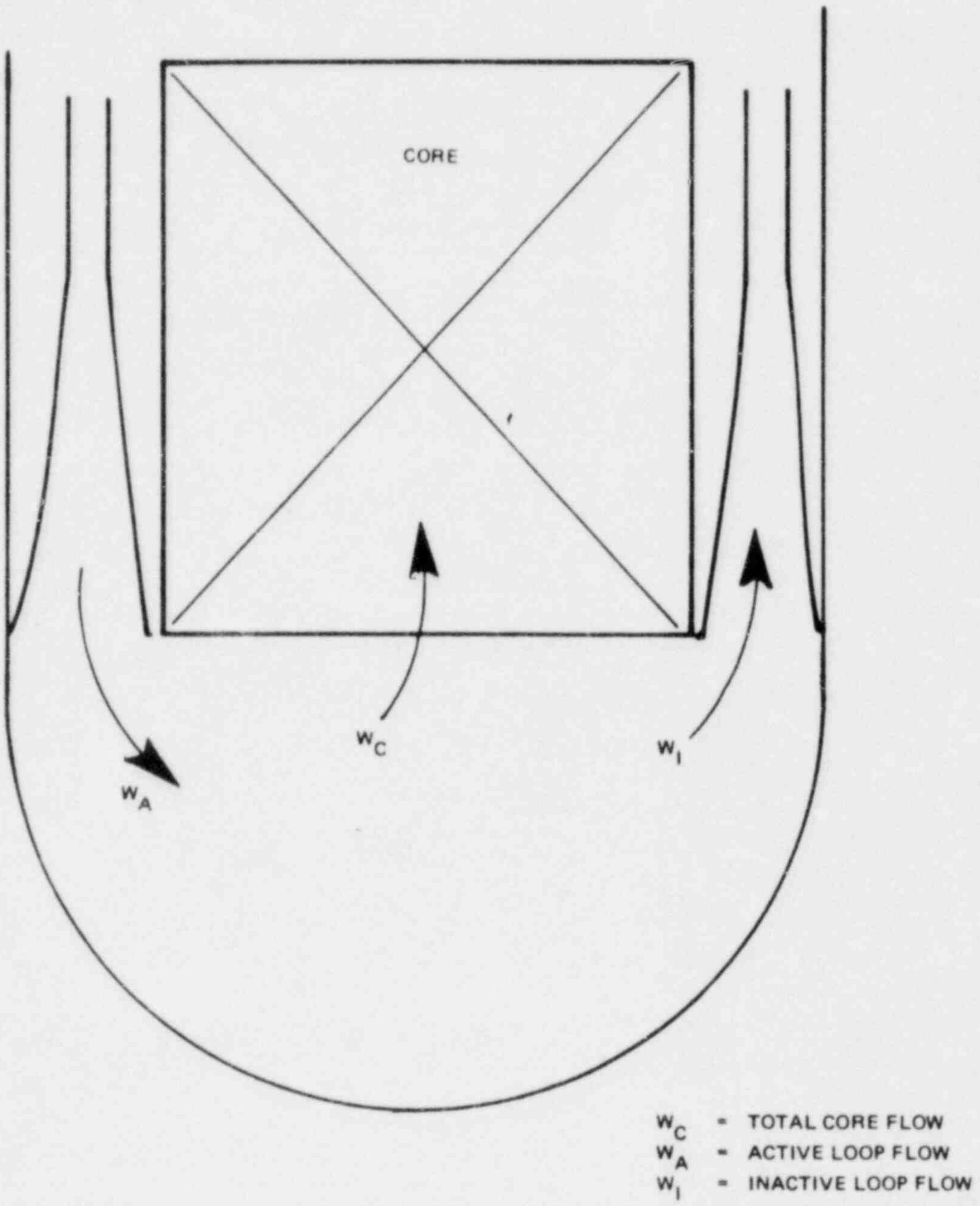

Figure 2-1. Illustration of Single Recirculation Loop Operation Flows

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/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 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 startup of an idle recirculation pump is not the limiting reactivity insertion transient. In addition, the restart of an idle pump would actually result in a neutron flux transient which would exceed the flow reference scram. The resulting transient with scram is expected to be less severe than the worst-case cold-water transient from rated power/flow.

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 increased subcooling of core inlet flow; 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 alarms which could occur during the course of the transient, the rod block system will stop rod withdrawal at a minimum critical power ratio which is higher than the fuel cladding integrity safety limit.

During single-loop operation, correction of the flow-biased Rod Block Monitor (RBM) equation (below) and the lower reactor power obtainable assures that the MCPR safety limit would not be violated during the postulated RWE.

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 when the active loop indicated flow is the same;

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 set point (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 operating MCPR 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 occurrence.

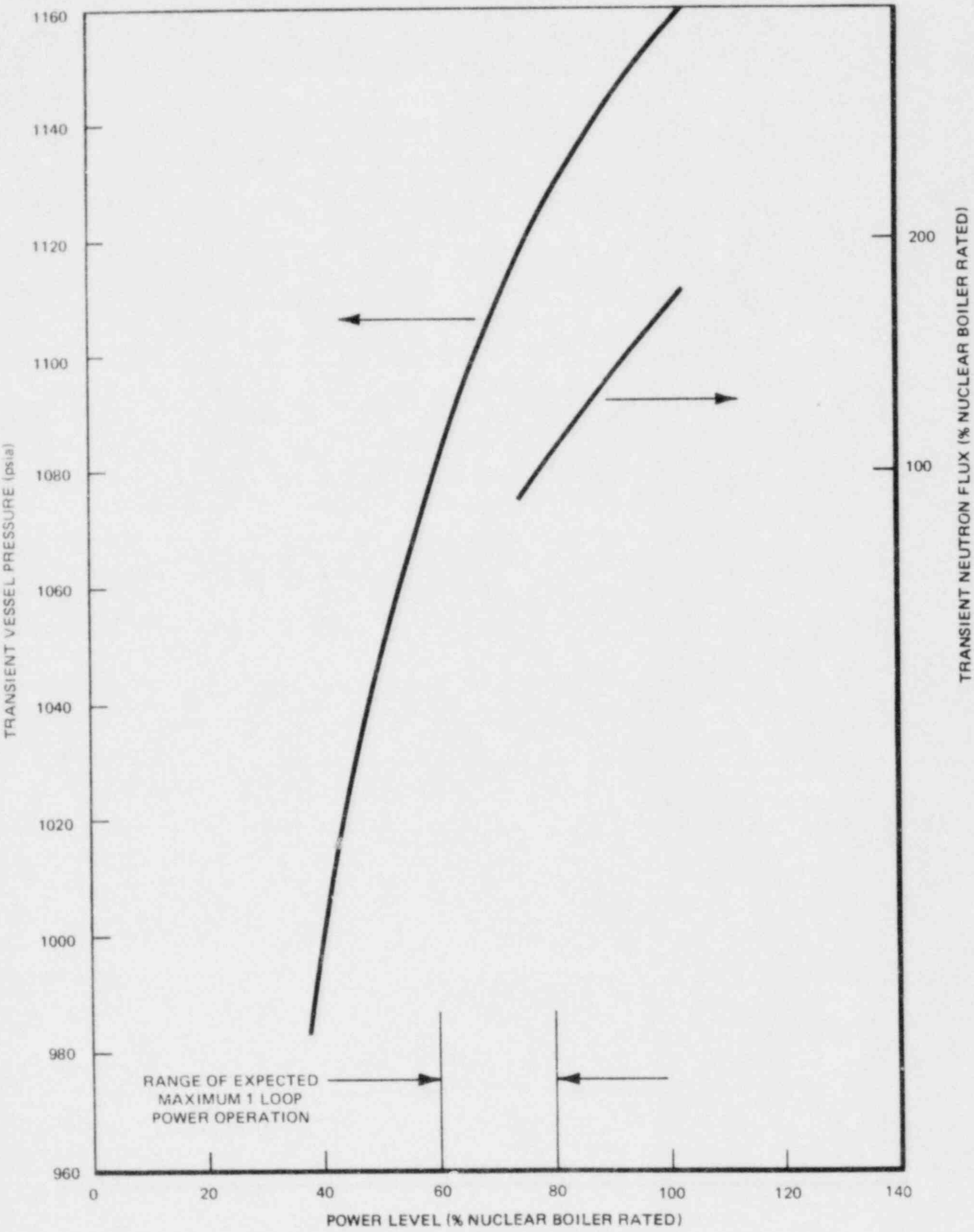

Figure 3-1. Main Turbine Trip with Bypass Manual Flow Control

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

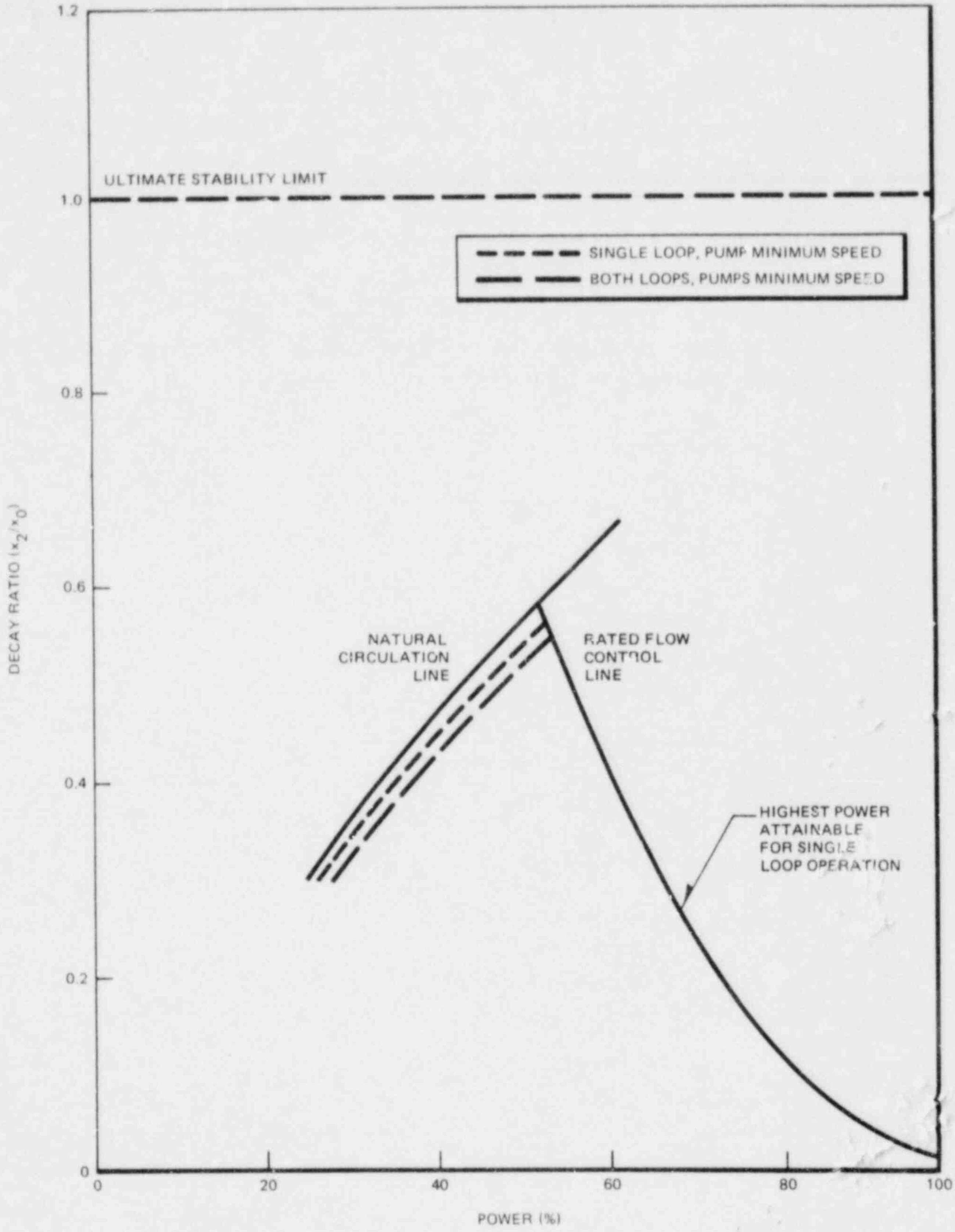

Figure 4-1. Decay Ratio versus Power Curve for Two-Loop and Single-Loop Operation

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 steam line break, 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 are conservatively applicable for one-pump operation.

5.1 LOSS-OF-COOLANT ANALYSIS

5.1.1 Break-Spectrum Analysis

A break spectrum analysis for each unit was performed using the model and assumptions given in Section II.A.7.3.2 of Reference 3. The suction and discharge break spectrum reflood times for one recirculation loop operation are compared to the standard previously performed two-loop operation in Figures 5-1 and 5-2, respectively, for Unit 1. Suction and discharge break spectrum reflood time comparisons for Unit 2 are shown in Figures 5-3 and 5-4. The uncovered time (reflood time minus recovery time) for the Unit 1 discharge and suction break spectrum and the Unit 2 discharge and suction break spectrum is compared in Figures 5-5, 5-6, 5-7, and 5-8, respectively.

For the Unit 1 standard two-loop analysis, the most limiting break was an 80% discharge Design Basis Accident (DBA) with a total uncovered time shown in Figure 5-5 and boiling transition times ranging from 9.7 to 10.3 seconds for the two fuel types.

For the Unit 1 single-loop analysis, a boiling transition time of [illegible] second is conservatively assumed for all breaks larger than 1.0 ft^2 and the reflooding times and total uncovered times are similar to the two-loop analysis. The most limiting break for single-loop analysis is the 86% discharge DBA which has a total uncovered time of 200.6 seconds. The single-loop reflooding time is 233.1 as opposed to a two-loop limiting reflooding time of 235 seconds.

For the Unit 2 standard two-loop break-spectrum analysis, the most limiting break is also an 80% discharge DBA with a total uncovered time shown in Figure 5-7 and a boiling transition time ranging from 9.7 to 10.3 seconds for the three fuel types.

For the Unit 2 single-loop analysis, a boiling transition time of 0.1 second is conservatively assumed for all breaks larger than 1.0 ft^2 and the reflooding times and total uncovered times are similar to the two-loc analysis. The most limiting break for single-loop analysis is also the 80% discharge DBA which has a total uncovered time of 215.3 seconds. The single-loop reflooding time is 248.6 as opposed to a two-loop reflooding time of 248.3 seconds.

5.1.2 Single-Loop Maximum Average Planar Linear Heat Generation Rate Determination

Since the reflooding time for the limiting break in single-loop operation for both Units 1 and 2 is similar to the reflooding time for two-loop operation, the procedure described in Section II.A.7.4 of Reference 3 is conservatively applicable.

Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) reduction factors were determined for the cases given in Table 5-1. The most limiting reduction factors for each fuel type are shown for both units 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.

The analyses were performed assuming the two recirculation manifolds are isolated from one another by closure of appropriate valves in the cross-tie (equalizer) line between the loops. 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 LPCI flow out of a postulated break in the idle suction line.

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. As this slight increase (50°F) in PCT is overwhelmingly offset by the decreased MAPLHGR (equivalent to 300° to 500°F PCT) for one-pump operation, the calculated PCT values for small breaks will be significantly below the 2200°F cladding temperature limit specified in 10CFR50.46.

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:

thermal power = 75% and core flow = 58%, and
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 a 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 Brunswick Steam Electric Plant, Units 1 and 2.

Table 5-1
MAPLHGR MULTIPLIER CASES

Unit	Fuel Type	Cases Calculated
1	8x8, 8x8R, and P8x8R	100% DBA Suction Break 100% DBA Discharge Break 80% DBA Discharge Break*
2	7x7, 8x8, 8x8R, and P8x8R	100% DBA Suction Break 80% DBA Discharge Break*

*Most limiting break.

Table 5-2

LIMITING MAPLHGR REDUCTION FACTORS

Unit	Fuel Type	Reduction Factor
1	8x8	0.85
	8x8R and P8x8R	0.85
2	7x7	0.84
	8x8	0.85
	8x8R and P8x8R	0.84

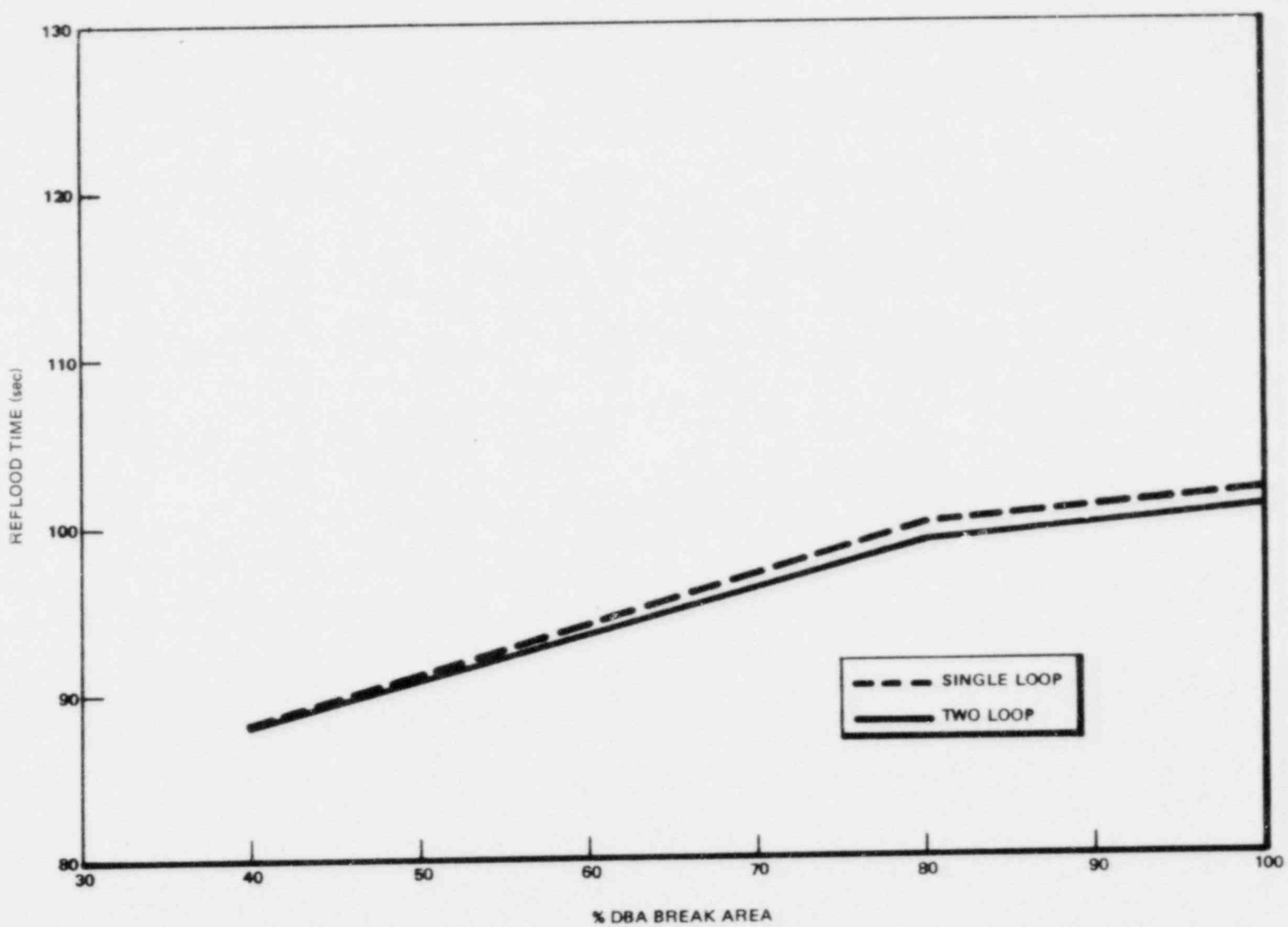

Figure 5-1. Brunswick 1 Suction Break Spectrum Reflood Times

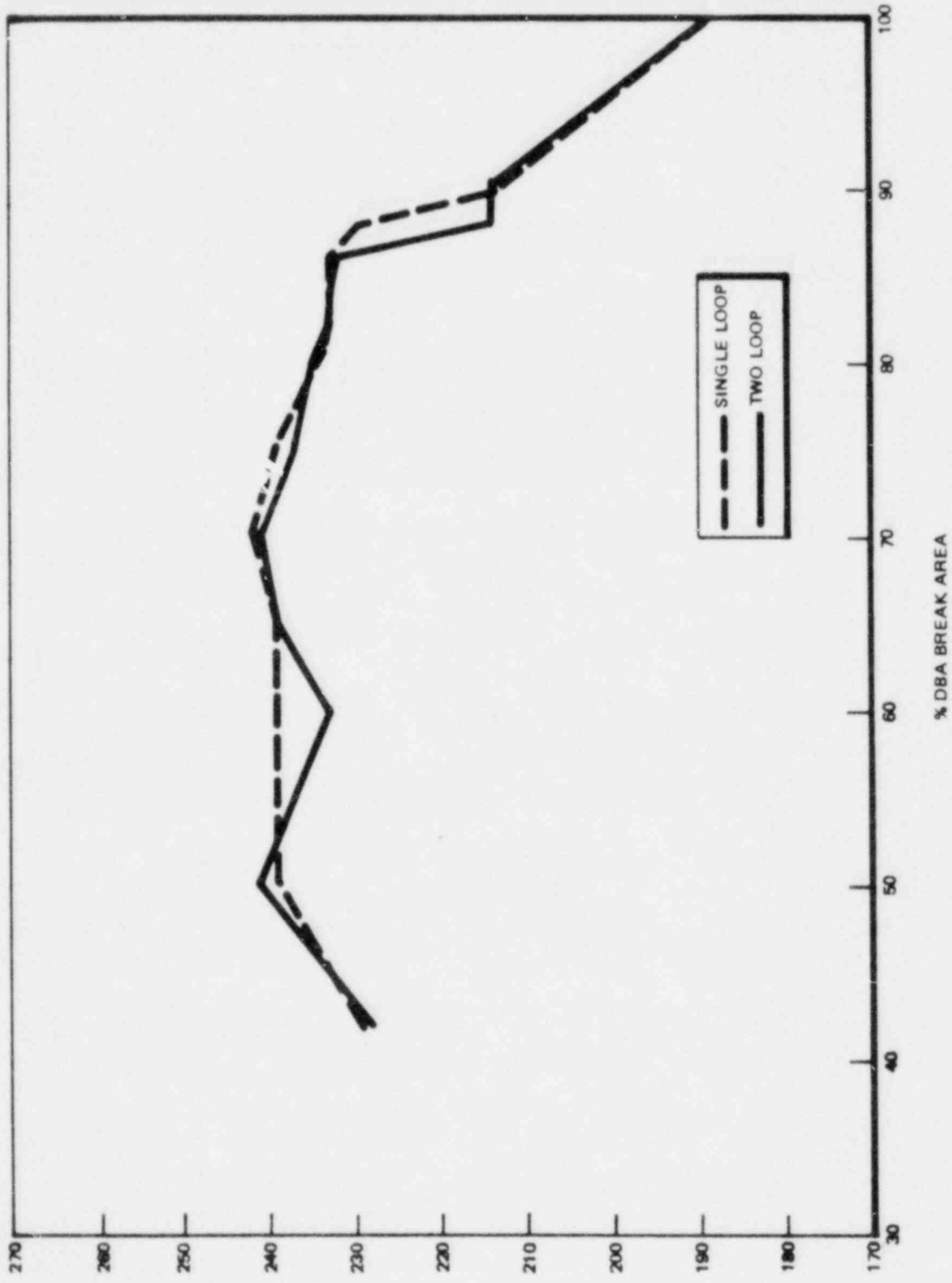

Figure 5-2. Brunswick 1 Discharge Break Spectrum Reflood Times

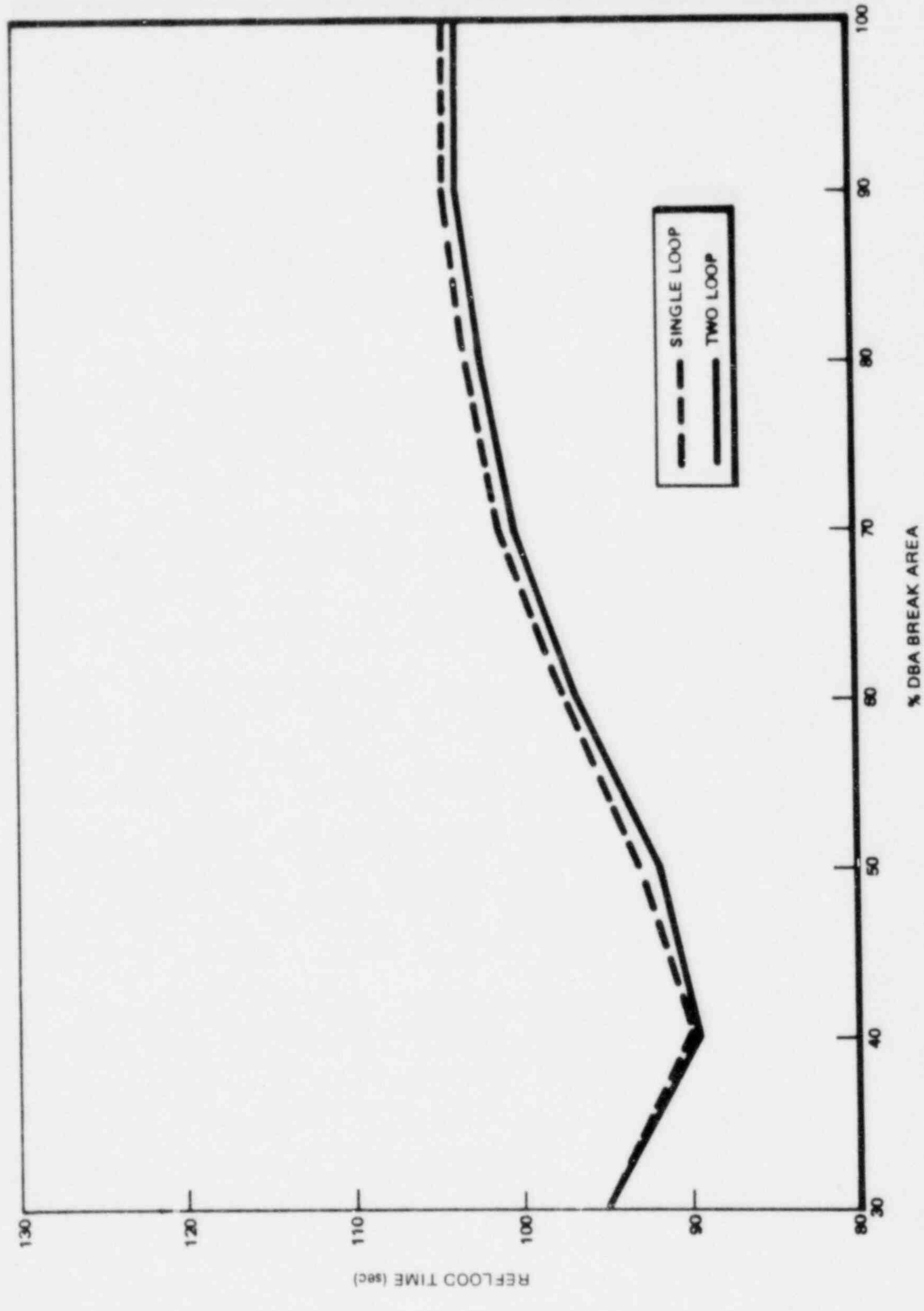

Figure 5-3. Brunswick 2 Suction Break Spectrum Reflood Times

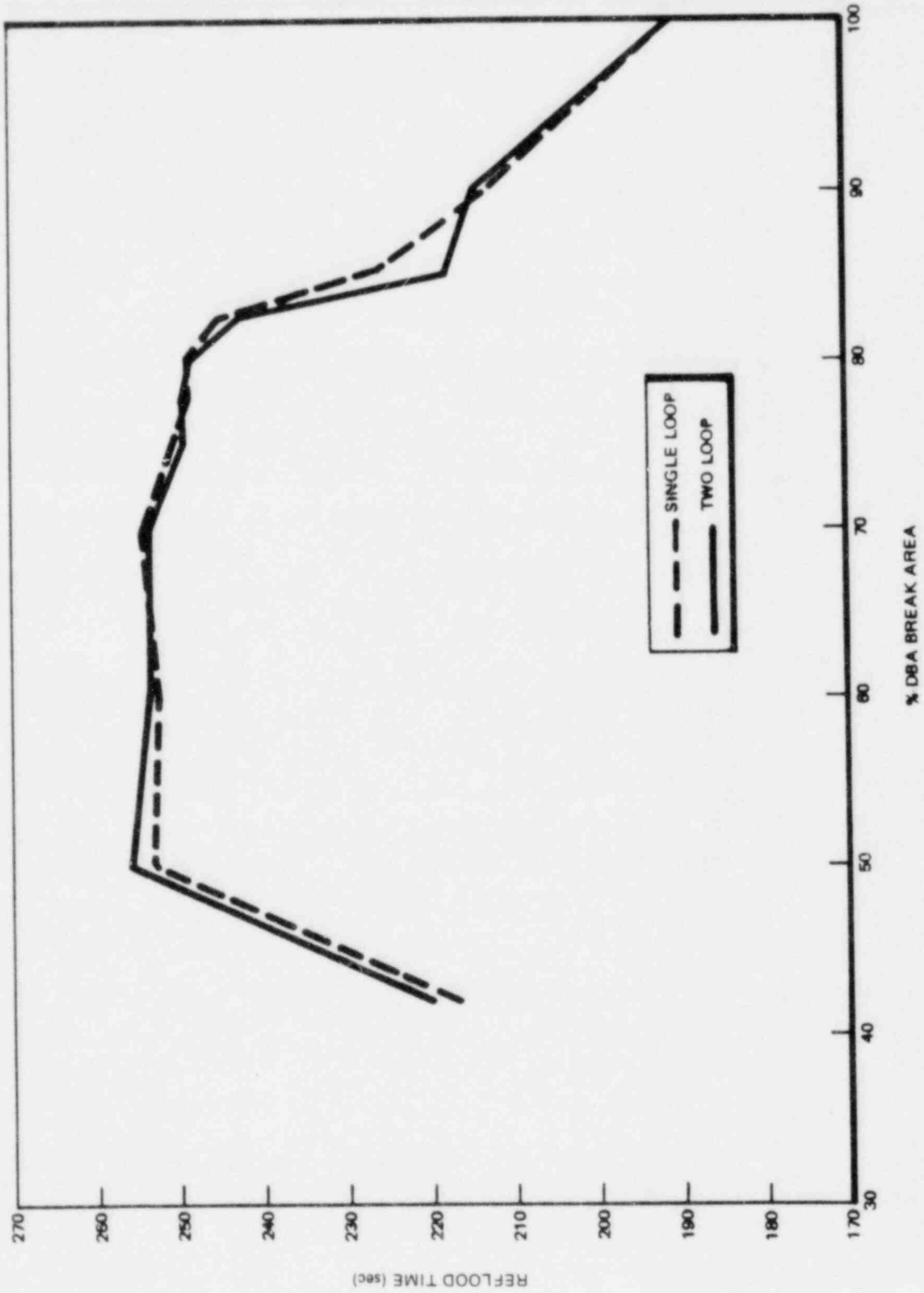

Figure 5-4. Brunswick 2 Discharge Break Spectrum Reflood Times

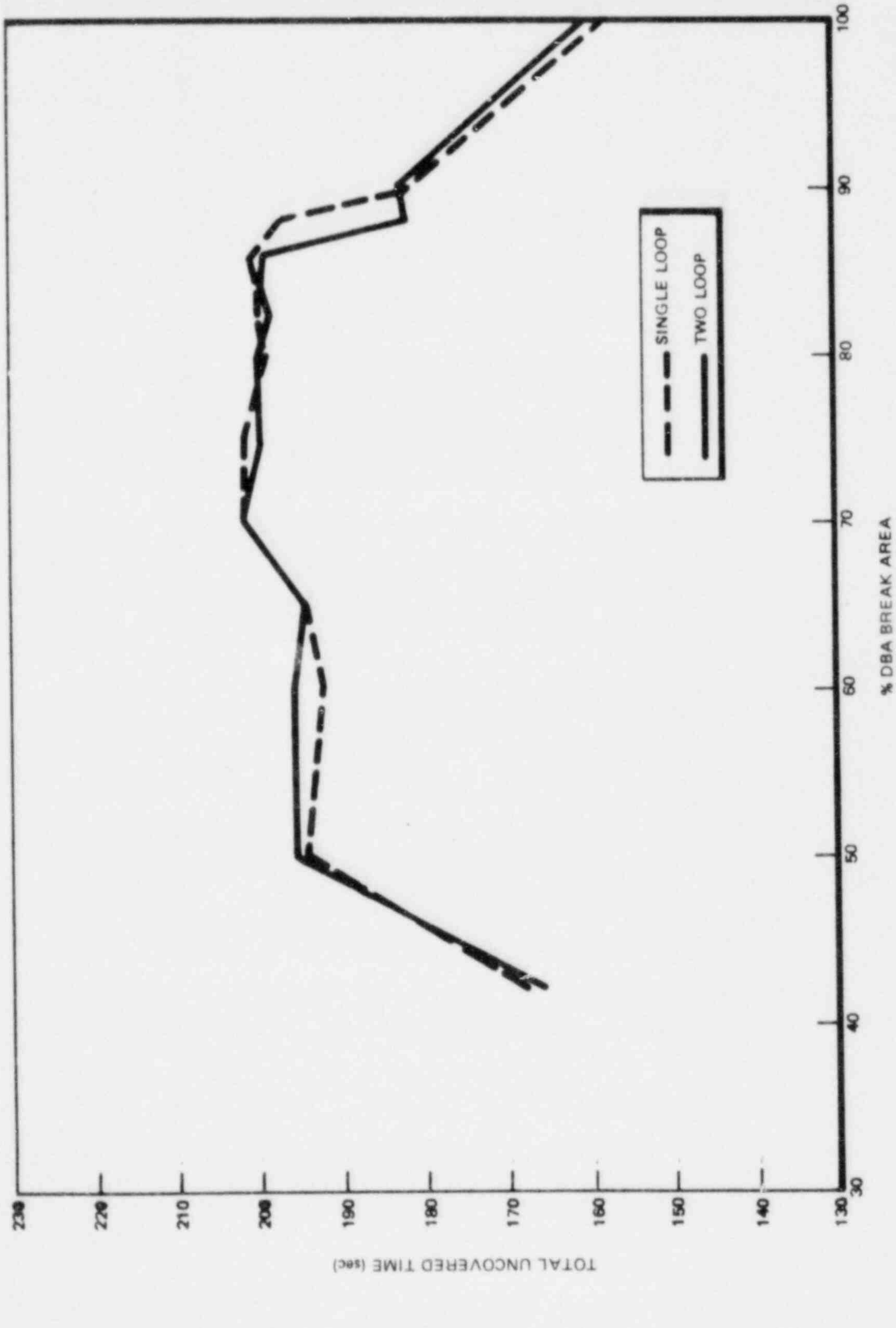

Figure 5-5. Brunswick 1 Discharge Break Spectrum Uncovered Times

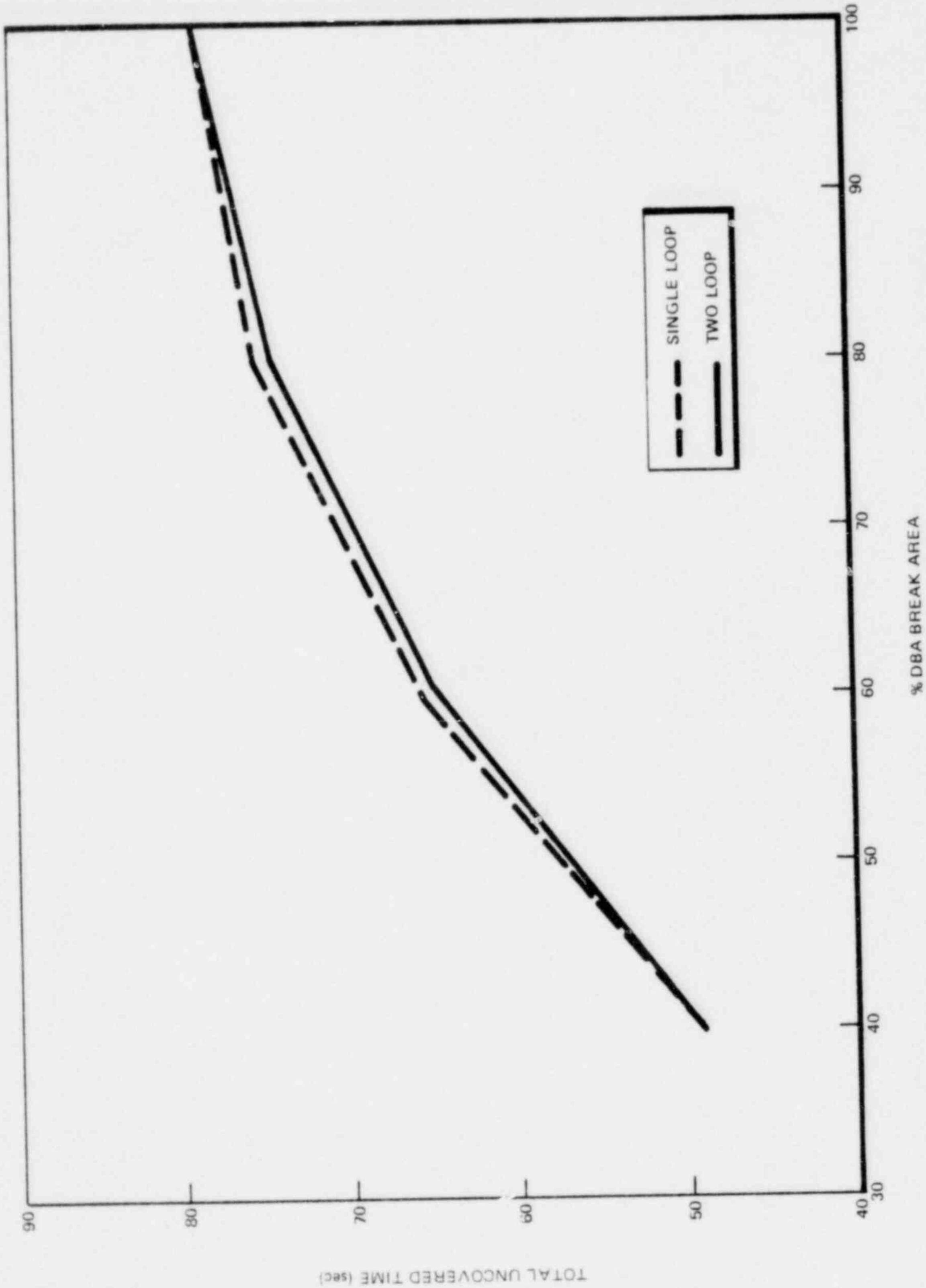

Figure 5-6. Brunswick 1 Suction Break Spectrum Uncovered Times

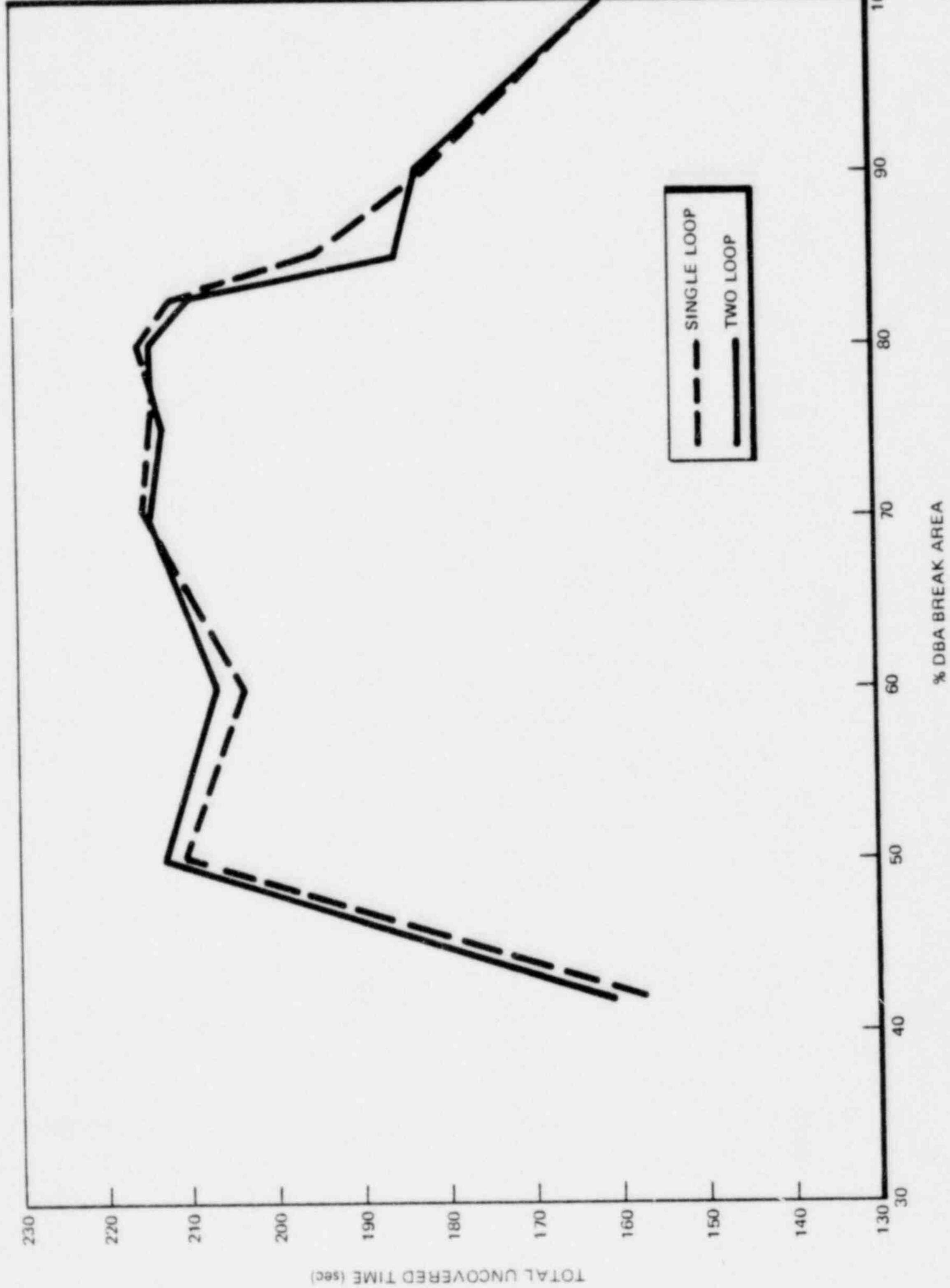

Figure 5-7. Brunswick 2 Discharge Break Spectrum Uncovered Times

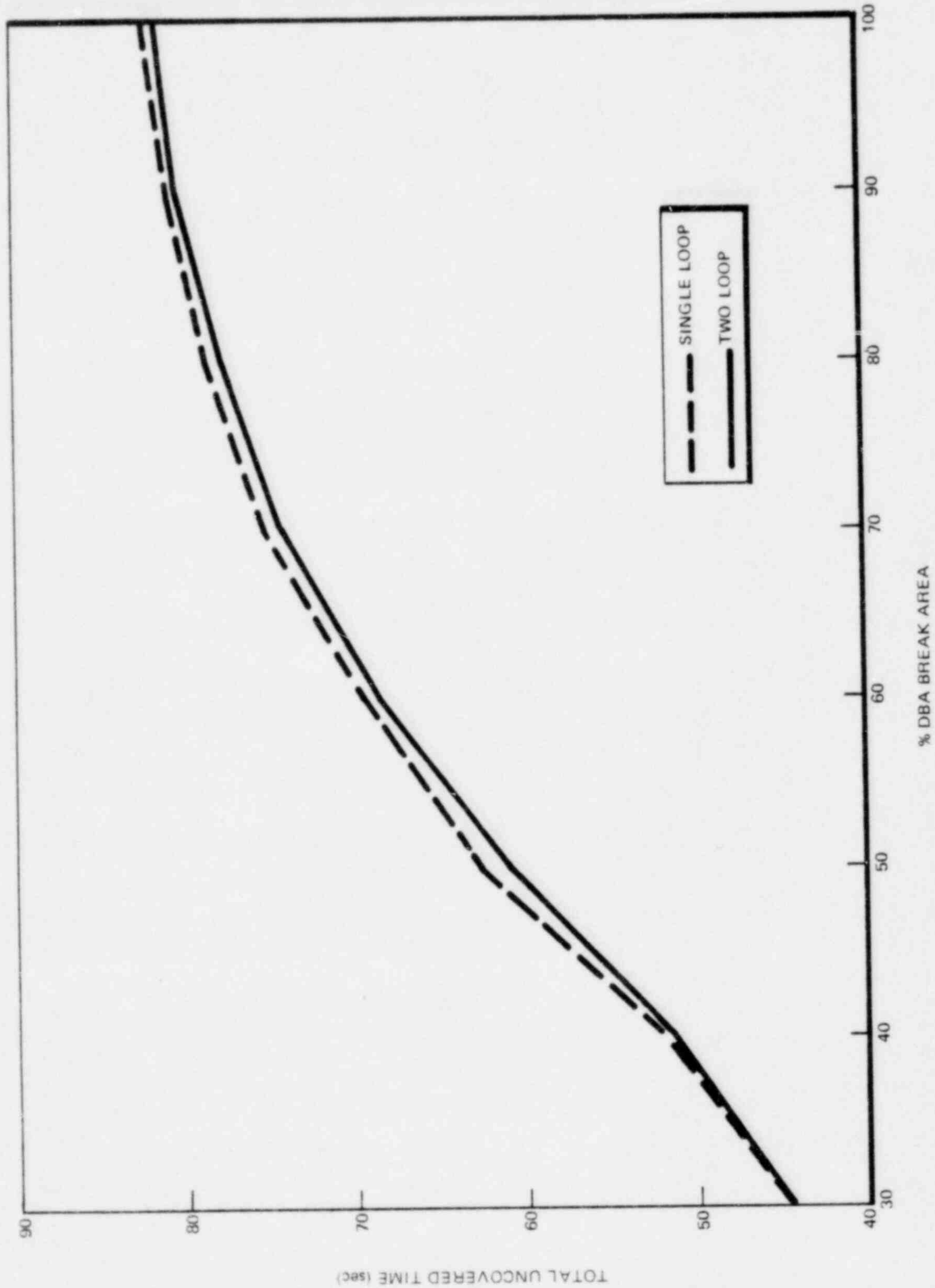

Figure 5-8. Brunswick 2 Suction Break Spectrum Uncovered Times

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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 No. TVA-BFNP-TS-117, O.E. Gray, III, to Harold R. Denton, September 15, 1978.

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