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SUBJECT: Application to amend Tech Specs App a to License DPR-49, Amend is for reactor startup & operation greater than 24-h w/one recirculation loop out of svc,Class III license change fee encl.

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APP 2

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Iowa Electric Light and Power Company

May 2, 1980
LDR-80-127

LARRY D. ROOT
ASSISTANT VICE PRESIDENT
NUCLEAR GENERATION

Mr. Harold Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Denton:

Transmitted herewith in accordance with the requirements of 10 CFR 50.59 and 50.90 is an application for amendment to Appendix A (Technical Specifications) to operating license DPR-49 for the Duane Arnold Energy Center (DAEC). The proposed amendment is for reactor startup and operation for greater than 24 hours with one recirculation loop out of service. Justification for this proposed change is also enclosed.

This application has been reviewed by the DAEC Operations Committee and the DAEC Safety Committee.

It has been determined that this is a Category III amendment and a payment of $4000.00 is herewith enclosed.

Three signed and 37 additional copies of this application are transmitted herewith. This application consisting of the foregoing letter and enclosures hereto, is true and accurate to the best of my knowledge and belief.

IOWA ELECTRIC LIGHT AND POWER COMPANY

Larry D. Root

Larry D. Root
Assistant Vice President
Nuclear Generation

Subscribed To And Sworn To Before Me On
This 2nd day of May, 1980.

Wilma J. Jewell

Notary Public In and For The
State of Iowa

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J. Van Sickle
T. Kevern (NRC)
File: A-117

Proposed Change RTS 119 to

The Duane Arnold Energy Center

Technical Specifications

The holders of License DPR-49 for the Duane Arnold Energy Center propose to amend Appendix A (Technical Specifications) to said license by deleting current pages and replacing them with the attached new proposed pages. A list of the affected pages is attached with the proposed new pages.

List of Affected Pages

1.1-2

1.1-3

3.2-16

3.2-17

3.6-7

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
16.C Power Transient To ensure that the Safety Limits established in Specification 1.1.A and 1.1.B are not exceeded, each required scram shall be initiated by its primary source signal. A Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the Primary Source Signal.	Where: S = Setting in percent of rated power (1,593 MWt) W = Recirculation loop flow in percent of rated flow. Rated recirculation loop flow is that recirculation loop flow which corresponds to 49×10^8 lb/hr core flow.
D. With irradiated fuel in the reactor vessel, the water level shall not be less than 12 in. above the top of the normal active fuel zone. Top of the active fuel zone is defined to be 344.5 inches above vessel zero (See Bases 3.2)	For a MFLPD greater than FRP, the APRM scram setpoint shall be: $S \leq (0.66W + 54) \frac{FRP}{MFLPD}$ for two recirculation loop operation and $S \leq (0.66W + 50.7) \frac{FRP}{MFLPD}$ for one recirculation loop operation. NOTE: These settings assume operation within the basic thermal design criteria. These criteria are LHGR $\leq$ 18.5 KW/ft (7x7 array) or 13.4 KW/ft (8x8 array) and MCPR $\geq$ values as indicated in Table 3.12-2 times K_f, where K_f is defined by Figure 3.12-1. Therefore, at full power, operation is not allowed with MFLPD greater than unity even if the scram setting is reduced. If it is determined that either of these design criteria is being violated during operation, action must be taken immediately to return to operation within these criteria.
	2. APRM High Flux Scram When in the REFUEL or STARTUP and HOT STANDBY MODE. The APRM scram shall be set at less than or equal to 15 percent of rated power.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
	3. APRM Rod Block When in Run Mode. For operation with MFLPD less than or equal to FRP the APRM Control Rod Block setpoint shall be as shown on Fig. 2.1-1 and shall be: $S \leq (0.66W + 42)$ The definitions used above for the APRM scram trip apply. For a MFLPD greater than FRP, the APRM Control Rod Block setpoint shall be: $S \leq (0.66W + 42)\frac{FRP}{MFLPD}$ for two recirculation loop operation, and $S \leq (0.66W + 38.7)\frac{FRP}{MFLPD}$ for one recirculation loop operation.
	4. IRM - the IRM scram shall be set at less than or equal to 120/125 of full scale.
B. Scram and Isolation on reactor low water level	$\geq$ 513.5 inches above vessel zero (+12" on level instruments)
C. Scram - turbine stop valve closure	$\leq$ 10 percent valve closure
D. Turbine control valve fast closure shall occur within 30 milliseconds of the start of turbine control valve fast closure.	

TABLE 3.2-C

Minimum No. of Operable Instrument Channels Per Trip System	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Action
2	APRM Upscale (Flow Biased)	for 2 recirc loop operation $\leq(0.66W + 42)\frac{FRP}{MFLPD}$ (2) for 1 recirc loop operation $\leq(0.66W + 38.7)\frac{FRP}{MFLPD}$ (2)	6 Inst. Channels	(1)
2	APRM Upscale (Not in Run Mode)	$\leq$12 indicated on scale	6 Inst. Channels	(1)
2	APRM Downscale	$\geq$5 indicated on scale	6 Inst. Channels	(1)
1 (7)	Rod Block Monitor (Flow Biased)	for 2 recirc loop operation $\leq(0.66W + 39)\frac{FRP}{MFLPD}$ (2) for 1 recirc loop operation $\leq(0.66W + 35.7)\frac{FRP}{MFLPD}$ (2)	2 Inst. Channels	(1)
1 (7)	Rod Block Monitor Downscale	$\geq$5 indicated on scale	2 Inst. Channels	(1)
2	IRM Downscale (3)	$\geq$5/125 full scale	6 Inst. Channels	(1)
2	IRM Detector not in Startup Position	(8)	6 Inst. Channels	(1)
2	IRM Upscale	$\leq$108/125	6 Inst. Channels	(1)
2 (5)	SRM Detector not in Startup Position	(4)	4 Inst. Channels	(1)
2 (5) (6)	SRM Upscale	$\leq 10^5$ counts/sec.	4 Inst. Channels	(1)

NOTES FOR TABLE 3.2-C

1. For the startup and run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function. The SRM and IRM blocks need not be operable in "Run" mode, and the APRM [except for APRM Upscale (Not in Run Mode)] and RBM rod blocks need not be operable in "Startup" mode. If the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days, the system shall be tripped. If the first column cannot be met for both trip systems, the systems shall be tripped.

2. W is the recirculation loop flow in percent of design. Trip level setting is in percent of rated power (1593 MWt). A ratio of FRP/MFLPD $<$ 1.0 is permitted at reduced power.

3. IRM downscale is bypassed when it is on its lowest range.

4. This function is bypassed when the count rate is $>$100 cps.

b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.

c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.

2. Whenever there is recirculation flow with the reactor in the Startup or Run mode, and one recirculation pump is operating, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than 10%.

F. Jet Pump Flow Mismatch

1. When both recirculation pumps are in steady state operation, the speed of the faster pump may not exceed 122% of the speed of the slower pump when core power is 80% or more of rated power or 135% of the speed of the slower pump when core power is below 80% of rated power.

2. If specification 3.6.F.1 cannot be met, one recirculation pump shall be tripped. The reactor may be started and operated with one recirculation loop out of service provided that:

a. A MAPLHGR multiplier of 0.65 is applied.

b. The power level is limited to a maximum of 50% of licensed power.

F. Jet Pump Flow Mismatch

1. Recirculation pump speeds shall be checked and logged at least once per day.

JUSTIFICATION FOR THE OPERATION OF

DUANE ARNOLD ENERGY CENTER

WITH ONE RECIRCULATION LOOP

OUT OF SERVICE

1.0 Introduction

The Technical Specifications for the Duane Arnold Energy Center (DAEC) (3.6.F.2) require that the plant be shutdown if an idle recirculation loop cannot be returned to service within 24 hours.

At approximately 5:00 AM, April 30, 1980, the "A" Loop Recirculation Pump Motor Generator (MG) Set tripped. Subsequent investigations indicate that a short to ground in the motor end of the MG Set exists requiring rewind of the motor prior to returning the "A" Loop to service. Present estimates indicate that a minimum of approximately three weeks are necessary to affect repairs.

The DAEC was returned to service after the spring refueling outage on April 18, 1980 which started February 9, 1980. The DAEC carries approximately 70% of Iowa Electric and partners load at this time of year. At 50% power capability, the DAEC would carry about 35% of the load. The effect of the DAEC not being operational is a severe impact to the rate payers.

In order to resume operation with one loop out of service, Iowa Electric has contracted GE to provide an analysis as outlined.

2.0 Special Operating Conditions for Single Loop Operation

In order to ensure operation of this derated condition is in accordance with the assumptions utilized by GE, Iowa Electric commits to the following conditions during normal operation.

1. Recirculation loop A recirculation pump is electrically disarmed and the motor is inoperable precluding operation of the pump or injection of a cold slug into the vessel.
2. The recirculation controls will be placed in the manual mode, thereby eliminating the need for control system analyses.
3. The settings for the rod block monitor, APRM rod block trip, and flow bias scram will be modified as necessary to provide for single loop operation. Technical Specifications are enclosed hereto.
4. Administrative Controls in addition to technical specifications restricting pump startup will prevent startup of the pump in the idle loop.
5. MAPLHGR restrictions, Figure 3.12-2 through 7, will result in a 35 percent reduction for all fuel.
6. The limitation on power level as described in FSAR Section 14.5.6.2 is 55 percent. Iowa Electric will further limit the power level to 50%.

3.0 MAPLHGR Adjustment Factor for DAEC

General Electric is performing analyses for single loop operation of DAEC. Preliminary evaluation of these calculations performed according to the procedure outlined in NEDO-20566-2 indicates that a multiplier of 0.86 should be applied to the MAPLHGR limits for single loop operation of the DAEC. Further, GE has performed a large number of single loop analyses for similar plants; in no case has a multiplier of less than 0.70 been required. Because DAEC does not have LPCI modification and because the limiting break is a suction line break, the single loop MAPLHGR multiplier is expected to be significantly better than for most other BWR's. Until the DAEC calculations can be verified (as required by 10.CFR.20), it is proposed that a multiplier of 0.65 be conservatively applied for single loop operation at the DAEC.

4.0 Other Considerations for Single-Loop Operation

Various conditions have been examined for the impact of single-loop operations. The following pages address several issues including:

A. One-pump seizure accident

B. Abnormal Operational Transients

1. Transients and Core Dynamics
2. Rod Withdrawal Error
3. APRM Trip Setting
4. K_f Curves

C. Stability Analysis

D. Thermal - Hydraulics

ONE-PUMP SEIZURE ACCIDENT

The pump seizure event is a very mild accident in relation to other accidents such as the 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 extremely rapidly; in the case of the 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 this provides a continuous core cooling mechanism. However, for the LOCA, complete flow stoppage occurs and the water level decreases due to loss-of-coolant resulting in uncovery of the reactor core and subsequent overheating 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 and specific analyses of the pump seizure are not required.

ABNORMAL OPERATIONAL TRANSIENTS

TRANSIENTS AND CORE DYNAMICS

Since operation with one recirculation loop results in a maximum power output which is 20 to 30% below that from which can be attained for two-pump operation, 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 for Reload/FSAR results bound both the thermal and overpressure consequences of one-loop operation. Figure 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 decreases transients are also bounded by the full power analysis. A single pump trip from one-loop operation is obviously 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 speed-up or introduction of colder water into the reactor vessel by events such as loss of feedwater heater. For the former, 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 set screws. This condition produces the maximum possible power increase and hence maximum ΔMCPR 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, and 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 increase in core power. This event is caused by positive reactivity insertion from 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 consequence from one-loop operation is bounded by previously submitted full power analysis. The maximum power level than can be attained on one-loop operation is only restricted by the MCPR and overpressure limits established from a full power analysis.

ROD WITHDRAWAL ERROR

The rod withdrawal error at rated power is given in reload licensing submittals. These analyses demonstrate that even if the operator ignores all indications and alarm which could occur during the course of the transient, the rod block system will stop rod withdrawal at a critical power ratio which is higher than the 1.07 safety limit. The MCPR requirement for one-pump operation will be equal to that for two-pump operation because the nuclear characteristics are independent of whether the core flow is attained by one- or two-pump operation. The only exceptions to this independence are possible flow asymmetries which might result from one-pump operation. Flow asymmetries are shown to be of no concern by tests conducted at Quad Cities. Under conditions of one-pump operation and

equalizer valve closed, flow was found to be uniform in each bundle. The DAEC does not have an equalizer line.

One-pump operation results in backflow through 8 of the 16 jet pumps while the flow is being supplied into the lower plenum from the 8 active jet pumps. Because of the backflow through the inactive jet pumps, the present rod block equation shown in the Technical Specification must be modified.

The procedure for modifying the rod block equation for one-pump operation is given in the following subsections.

a. The two-pump rod block equation in the existing Technical Specifications is of the form:

$$RB = (mW + K)\% \qquad (1)$$

where

RB = power at rod block in %

m = flow reference slope for the rod block monitor (RBM)

W = drive flow in % of rated

K = power at rod block in % when W = 0.

For the case of top level rod block at 100% flow, denoted RB_{100}:

$$RB_{100} = m(100) + K$$

or

$$K = RB_{100} - m(100)$$

Substituting for K in Equation 1, the two pump equation becomes:

$$RB = mW + [RB_{100} - m(100)] \qquad (2)$$

b. Next, the core flow (F_c) versus drive flow (W) curves are determined for the two-pump and one-pump cases. For the two-pump case the core flow and drive flow are derived by measuring the differential pressures in the jet pumps and recirculation loop, respectively. Core flow for one pump operation must be corrected for the backflow through the inactive

jet pumps thus:

Actual core flow (one pump) = Active jet pump flow - inactive jet pump flow.

Both the active and inactive flows are derived from the jet pump differential pressures. The drive flow is derived from the differential pressure measurement in the active recirculation loop. These two curves are plotted from a BWR data in Figure 2. The maximum difference between the one-pump and two-pump core flow is determined graphically. This occurs at about 35% drive flow which is denoted W.

c. Next, a horizontal line is drawn from the 35% drive flow point on the one pump curve to the two pump curve and the corresponding flow, W_2, is determined. Thus, $\Delta W = W_1 - W_2$.

The rod block equation corrected for one pump flow is:

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

where

$$\Delta RB = RB_1 - RB_2 = m\Delta W$$

$$RB = mW + RB_{100} - m(100 + \Delta W) \qquad (3)$$

d. For DAEC application, the constants from the Technical Specification are:

$m = 0.66$

$RB_{100} = 108$

From Figure 2:

$$\Delta W = W_1 - W_2 = 35 - 30 = 5$$

Evaluating in Equation 3, the one-pump rod block equation becomes:

$$RB = 0.66W + 108 - 0.66(100+5) = 0.66W + 38.7 \qquad (4)$$

This line is depicted in Figure 2 as the future corrected rod block line for one-pump operation.

APRM TRIP SETTING

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.

K_f CURVE

For single recirculation loop operation, the K_f curve contains sufficient conservatism to provide operational limits such that the fuel integrity safety limit is not violated for abnormal operational events.

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. 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. The core stability along the forced circulation, rated rod pattern line for single loop operation is the same as that for both loops operable except that rated power is not attainable. Hence, the core is limited to maximum power for single pump operation and only manual flow control should be used. This is illustrated in Figure 3.

THERMAL-HYDRAULICS

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. The only exception is the core total flow. The standard deviation for this quantity from Table 4-2 is 2.5%. For single-loop operation this value may increase to about 6% of rated core flow.* The 3.5% increase in core total flow uncertainty corresponds to an increase in the safety limit of about 0.004% which can be neglected.

The steady-state operating MCPR with single-loop operation will be conservatively established by multiplying the K_f factor to the rated flow MCPR limit. This ensures that the 99.9% statistical limit requirement is always satisfied.

*See Appendix A for justification.

SHORT TERM LESSONS LEARNED

Iowa Electric was requested by the NRC to evaluate single-loop operations in light of the Short-Term Lessons Learned from TMI. Iowa Electric and General Electric have conducted a review of the Short-Term Lessons Learned and have determined that there is no unique consideration which would affect any actions taken in accordance with the Short-Term Lessons Learned.

Iowa Electric will conduct training with all Control Room operators prior to those personnel operating the plant in single loop. This training will consist of review of appropriate Technical Specifications, core limits, precautions, surveillance tests, and bases for single loop operation. As the plant is normally operated in local manual recirculation control at less than 50% power the operations are not unique and specific training is not required.

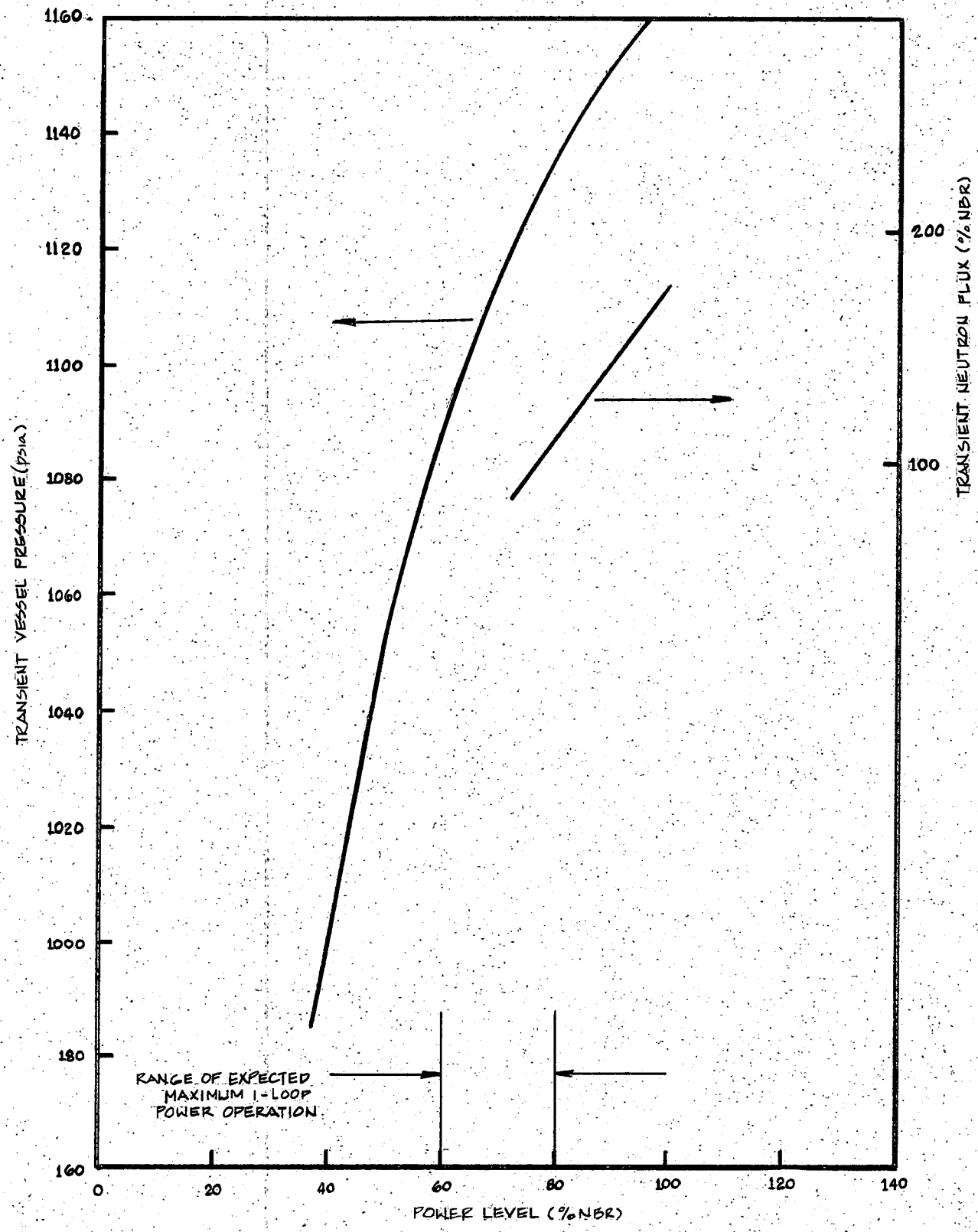

FIGURE 1. MAIN TURBINE TRIP WITH BYPASS MANUAL FLOW CONTROL

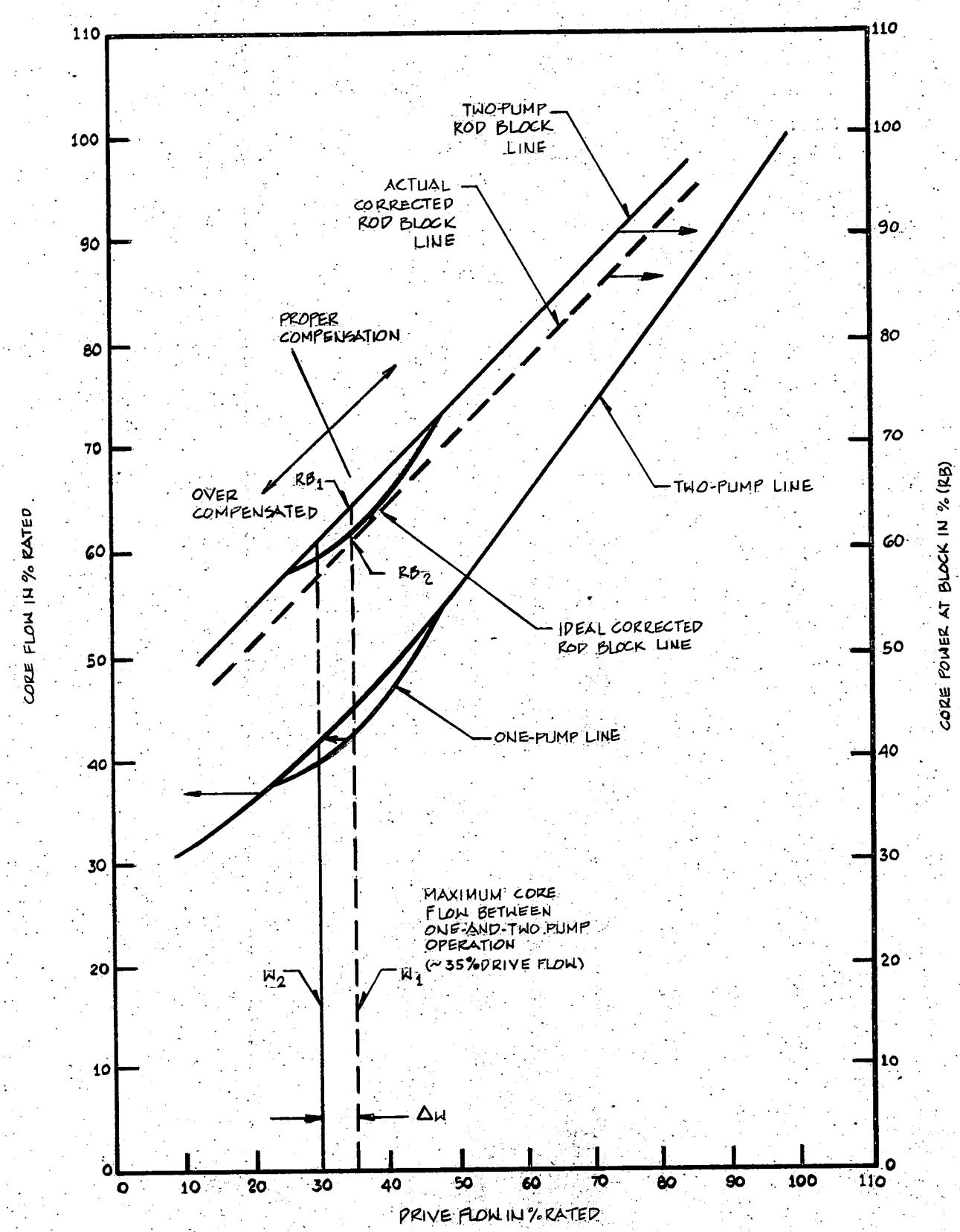

FIGURE 2. CORE FLOW VERSUS DRIVE FLOW FOR ONE-AND-TWO-PUMP OPERATION

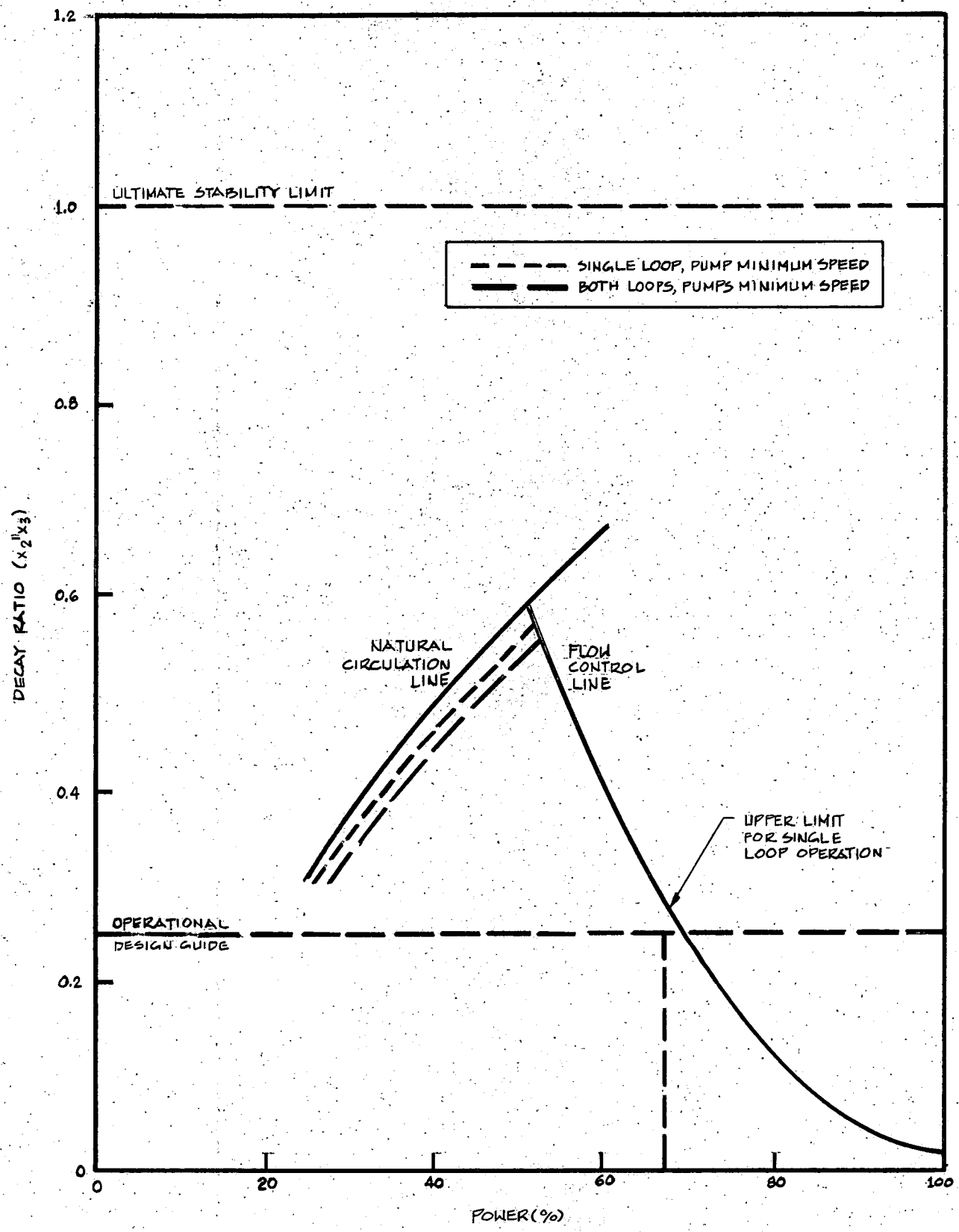

FIGURE 3. DECAY RATIO VERSUS POWER CURVE FOR TWO-LOOP AND SINGLE-LOOP OPERATION

APPENDIX A

UNCERTAINTIES IN TOTAL CORE FLOW FOR SINGLE LOOP OPERATION

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 and the total core flow is the sum of the indicated loop flows. However, for single loop operation, the inactive jet pumps will be backflowing, so 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 in reverse flow than forward flow, and the measurement of reverse flow must be modified to account for this fact.

For single loop operation the total core flow should be measured 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 and "Loop Indicated Flow" means the flow indicated by the jet pump "single-tap" loop flow summers and indicators, which are set up 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.* If a more exact, less conservative core flow is required, special in-reactor calibration tests could be made. Such calibration tests may involve calibrating core support plate ΔP versus core flow during two pump operation along the 100% flow control line, then 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, along with the loop flow indicator readings.

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*Note: The expected value of the "C" coefficient is ~0.89.

2. CORE FLOW UNCERTAINTY ANALYSIS

The uncertainty analysis procedure used to establish the core flow uncertainty for one pump operation is basically the same as for two pump operation, except for some extensions. The core flow uncertainty analysis is described in References 1 and 2. The analysis of one pump core flow uncertainty can be summarized as follows:

a. During one pump operation, the core flow is measured by the following formula:

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

$$= \left(\sum_{i=1}^{10} \sqrt{\frac{\Delta P_{ST_i}}{K_{forward}}}\right) - \sqrt{\frac{K_{forward}}{K_{reverse}}} \left(\sum_{i=11}^{20} \sqrt{\frac{\Delta P_{ST_i}}{K_{forward}}}\right)$$

where ΔP_{ST_i} is the "single tap" differential pressure for jet pump "i".

The constant "C" is required to modify the inactive loop flow indication since the jet pump diffuser flow coefficient is different for reverse flow compared with the forward flow coefficient used for the core flow instrumentation calibration.

b. The core flow uncertainty analysis must now account for the uncertainty in "C". The value of "C" has been determined analytically, using a conservative bounding analysis: therefore, the core flow input to the process computer during one pump operation has a conservative bias, since "C" was analyzed in a conservative manner. However, the following uncertainty analysis is based on the uncertainty in the true (or nominal) value of "C," not the uncertainty in the conservative value of "C" used in the reactor flow measurement.

"C" can be defined as:

$$C = \sqrt{\frac{K_{forward}}{K_{reverse}}}$$

where:

$K_{forward}$ = The forward flow loss coefficient resulting from in-reactor calibration tests assumed for the analytical derivation of "C."

Note: $K \equiv (flow)^2 \quad (\Delta P)$

$K_{reverse}$ = The loss coefficient calculated for reverse flow.

Combining the uncertainties in $K_{forward}$ and $K_{reverse}$, it can be shown that

$$\sigma_c^2 = \frac{1}{4}\left[\sigma^2_{K_{forward}} + \sigma^2_{K_{reverse}}\right]$$

Bounding values are $\sigma_{K_{forward}} \leq 2.5\%$ and $\sigma_{K_{reverse}} \leq 6.4\%$, thus:

$$\sigma_c = 3.4\%$$

c. Now the effect of this reverse flow coefficient uncertainty must be related to total core flow uncertainty. Assuming that 33%* of the flow in the active (forward flowing) jet pumps backflows throughthe inactive pumps, it can be shown that:

$$\sigma^2_{W_T} = \sigma^2_{W_A} + \left(\frac{0.33}{1-0.33}\right)^2 \sigma_c^2$$

*Note: This value can vary from about 20% to 30%, depending on plant type and operating conditions. 33% is a conservative bounding value.

where:

σ_{W_T} = the uncertainty in the total core flow.

σ_{W_A} = the uncertainty in the active loop flow.

The flow uncertainty in the active jet pumps during single loop operation (σ_{W_A}) is presently analyzed to be <3.5%. To produce a conservative, bounding analysis, σ_{W_A} = 4.0% was used in this calculation. Then,

$$\sigma^2_{W_T} = (4.0\%)^2 + \left(\frac{0.33}{1-0.33}\right)^2 (3.4\%)^2 = (4.34\%)^2$$

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:

$$\sigma^2_{\substack{\text{active}\\ \text{coolant}}} = (4.34\%)^2 + \left(\frac{0.12}{1-0.12}\right)^2 (4.1\%)^2 = (4.38\%)^2$$

This verifies the assumption of core flow uncertainties of 6%. Actually, the core flow accuracy is expected to be much better, as shown above.

In summary, core flow during one pump operation is measured in a conservative way, its uncertainty has been conservatively evaluated, and the net effect on MCPR is insignificant.

REFERENCES - Appendix A

1. Letter to Walter R. Butler (AEC/NRC). Subject: Response to the Third Set of AEC Questions on the General Electric Licensing Topical Reports NEDO-10958 and NEDE-10958, "General Electric BWR Thermal Analysis (GETAB): Data, Correlation and Design Application," July 11, 1974.

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