Public Service
Electric and Gas
Company

Steven E. Miltenberger Public Service Electric and Gas Company P.O. Box 236, Hancocks Bridge, NJ 08038 609-339-4199

Vice President and Chief Nuclear Officer

October 19, 1988

NLR-N88171

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Gentlemen:

EMERGENCY LICENSE AMENDMENT REQUEST
TECHNICAL SPECIFICATION 3/4.2.3
SALEM GENERATING STATION
UNIT NO. 2
FACILITY OPERATING LICENSE DPR-75
DOCKET NO. 50-311

Public Service Electric and Gas Company (PSE&G) hereby submits a request to amend Appendix A of Facility Operating License DPR-75 in accordance with 10 CFR 50.90. This request for amendment would reduce the correction for flow measurement uncertainty from 3.5% to 2.2%. The justification for the reduction is provided in WCAP-11579 which was prepared in support of Salem Unit No. 1 and 2 RTD Bypass removal modifications (Amendment Nos. 84 and 56, respectively). WCAP-11579 was previously submitted for NRC review under PSE&G's letter NLR-N87157 dated September 17, 1987.

It has been determined that the proposed amendment does not involve a significant hazards consideration pursuant to 10 CFR 50.92. A description of the amendment request is provided as Enclosure 1. Enclosure 2 provides Salem Unit 2 secondary side calorimetric instrument uncertainties for comparison with the values assumed in WCAP-11579. Enclosure 3 provides marked up Salem Unit No. 2 Technical Specification pages. Enclosure 4 provides the retyped and revised Salem Unit 2 Technical Specification pages.

PSE&G has evaluated this request pursuant to 10 CFR 170.21 and has determined that a license amendment application fee is required. A check for $150.00 is enclosed in payment of this fee. In accordance with 10 CFR 50.91(b)(1), a copy of this amendment request has been sent to the State of New Jersey.

A001
1/37
w/check $150
#2418

Pursuant to 10 CFR 50.4(b)(2)(ii), this submittal includes one (1) signed original and thirty-seven (37) copies. Should you have any questions regarding this submittal, please do not hesitate to contact us.

Sincerely,

Steven E Miltenberger

Enclosures

C Mr. J. C. Stone
Licensing Project Manager

Mr. R. W. Borchardt
Senior Resident Inspector

Mr. W. T. Russell, Administrator
Region I

Ms. J. Moon, Interim Chief
New Jersey Department of Environmental Protection
Division of Environmental Quality
Bureau of Nuclear Engineering
CN 415
Trenton, NJ 08625

REF: LCR 88-13

STATE OF NEW JERSEY)
) SS.
COUNTY OF SALEM)

Steven E. Miltenberger, being duly sworn according to law deposes and says:

I am Vice President and Chief Nuclear Officer of Public Service Electric and Gas Company, and as such, I find the matters set forth in our letter dated October 19, 1988 , concerning the Salem Generating Station, Unit No. 2, are true to the best of my knowledge, information and belief.

Steven E Miltenberger

Subscribed and Sworn to before me
this 19th day of October, 1988

Eileen M. Ochs
Notary Public of New Jersey

EILEEN M. OCHS
NOTARY PUBLIC OF NEW JERSEY
My Commission expires on My Commission Expires July 16, 1992

ENCLOSURE 1

REQUEST FOR AMENDMENT
SALEM GENERATING STATION UNIT NO. 2
FACILITY OPERATING LICENSE DPR-75
DOCKET NO. 50-311

DESCRIPTION OF CHANGE

Specification 3/4.2.3, RCS Flow And R, and Figure 3.2-3 establish a minimum allowable RCS flow rate of 36.1E4 gpm. This flow rate is based on a thermal design flow rate of 8.73E4 gpm per loop with a 3.5% correction for flow measurement uncertainties. The requested change reduces the RCS Flow uncertainty correction to 2.2% and revises Figure 3.2-3 to reflect the appropriate adjustments to the minimum flow limit. Miscellaneous typographical corrections are also made.

The reduced uncertainty value was developed as part of the RTD Bypass removal modification. The supporting analysis is contained in WCAP-11579. WCAP-11579 is a Westinghouse proprietary document which was transmitted to the NRC by PSE&G letter NLR-87157 dated September 17, 1987. This transmittal also included the appropriate request for withholding of proprietary information as provided for in 10CFR2.790.

BASIS FOR CHANGE

PSE&G letter NLR-87078 dated May 5, 1987, PSE&G submitted license amendment request 87-05 to revise the Salem Unit 1 and 2 Technical Specifications to reflect modifications to the RCS narrow range temperature indication system. This modification consisted of removing the existing RTD Bypass Loop and manifolds and replacement of the existing RTDs with thermowell RTDs. A modified scoop-thermowell design was also incorporated. The final temperature signal for a given loop was then developed from the averaged output of three RTDs instead of a single RTD as with the previous design.

To determine the impact of the RTD Bypass Loop removal on the Salem temperature related control and protection functions, Westinghouse performed instrument uncertainty calculations which utilized the latest available information on plant installed instrumentation and the modified scoop-thermowell design. To perform the calculations, Westinghouse employed a methodology similar to that used to justify RTD Bypass Loop elimination at McGuire, Catawba, Byron, and Braidwood. This methodology calculates all of the various uncertainties and values pertaining to a reactor trip. The basic method of error combination was the same as that used in the V.I. Sumner setpoint study, also performed by Westinghouse, with the exception that adjustments were made to reflect the use of 3 hot leg RTDs and the removal of the bypass piping. The details and results of this evaluation are provided in WCAP-11579.

The present RCS Flow correction (3.5%) reflects the uncertainties associated with the previous (i.e. RTD Bypass Loop) RCS narrow range temperature monitoring system. Thermal design flow (TDF) rate is a key parameter established during the calculation of the design DNBR value. Once established, the thermal design flow rate must then be corrected for measurement uncertainties to establish a minimum allowable flow rate for safe plant operation. The minimum allowable flow (MAF) rate shown in Figure 3.2-3 is determined by applying the uncertainty value in the following fashion;

$$MAF = TDF * (1 + .035)$$

The end result is that Figure 3.2-3 may be used without any additional correction to the RCS flow rate determined by performance of the calorimetric.

Included in WCAP-11579 was a review of the impact of the modified narrow range temperature monitoring system on RCS Flow Calorimetric uncertainties. The details of this analysis, including the assumptions relative to calorimetric instrument uncertainties are included in Section 3.0 of WCAP-11579. It was concluded that the RCS flow calorimetric uncertainty could be reduced to 2.2% from the previously calculated value of 3.5% largely due to the use of 3 RTDs for the development of the hot leg temperature signal. Because three RTDs are used to measure hot leg temperature instead of a single measurement, the error associated with the measurement is reduced to one over the square root of three compared to a single RTD. PSE&G reviewed the calorimetric instrument uncertainty assumptions contained in WCAP-11579 and discussed the differences between Salem Unit 2 and the WCAP with Westinghouse. Applicable portions of the WCAP uncertainty calculation were reanalyzed using Salem Unit 2 uncertainty values to assure that the 2.2% uncertainty value bounded the Salem Unit No. 2 design. Calculations of the Salem Unit 2 instrument uncertainties are provided as Enclosure 2 for comparison with the WCAP-11579 values.

The 3.5% correction presently included in Figure 3.2-3 establishes a reference point from which the relative magnitude of the safety margin associated with the value of TDF can be inferred. PSE&G does not propose to revise the value of TDF which has been used in the DNBR calculation. As a result the present safety margin associated with the TDF value has been increased as a result of the installation of the modified narrow range temperature monitoring system. The requested change applies only to the correction as applied to Figure 3.2-3 and the resultant minimum allowable RCS Flow Rate. The net result of the reduction in minimum allowable flow is to establish a new reference point while maintaining the magnitude of the existing safety margin associated with the value of TDF.

SIGNIFICANT HAZARDS EVALUATION

The standards used to arrive at a determination that a request for amendment involves no significant hazards consideration are included in the Commission's regulations, 10CFR50.92. These regulations state that no significant hazards considerations are involved if the operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Each standard is discussed as follows:

1. As stated previously, the reduction in the uncertainty value is attributed to the reduced error associated with the modified RCS narrow range temperature monitoring system. The Chapter 15 accident analyses impacted by this modification were previously reviewed and approved by the NRC as amendments 84 and 56 to the Salem Unit 1 and 2 licenses, respectively.

   The requested change does not result in a reduction of the RCS thermal design flow which was assumed for the purpose of accident analysis. The requested change therefore does not result in a value of DNBR which is less than the minimum design DNBR value identified in the Updated Final Safety Analysis Report.

   Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The correction is based on an analysis of flow measurement uncertainties. The correction does not affect any process variable which inputs to a process control or reactor protection system control function. Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any previously evaluated.

3. An RCS Flow uncertainty error of 3.5% was assumed for the purpose of calculating a minimum allowable RCS flow rate for safe plant operation. The uncertainty correction provides a reference point from which the relative magnitude of the safety margin between measured flow rate and design thermal flow rate can be inferred. WCAP-11579 demonstrates that the total uncertainty associated with the modified RCS narrow range temperature monitoring system could be reduced to a conservative value of 2.2% from the existing value of 3.5%. If the uncertainty is not reduced then a net increase in the margin of safety associated with the present thermal design flow value is achieved.

   As discussed previously, PSE&G does not intend to revise the value of thermal design flow used in the DNBR analysis. The requested change is only being applied to Figure 3.2-3 and the resultant value of minimum allowable RCS flow. The net result of the requested change is to redefine the reference point but maintain the magnitude of the existing margin of safety.

   It is therefore concluded that operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

PSE&G concludes that this amendment is not likely to involve a significant hazards consideration since it does not involve a process or plant operating condition which is different from or less conservative than any previously evaluated. The change involves reducing the applied RCS flow correction as a result of reduced instrument error associated the improved RCS narrow range temperature monitoring system.

<u>Bases for Emergency Circumstance</u>

During required Salem Unit 2 steam generator tube eddy current testing and the identification of numerous defective tubes, PSE&G committed to plugging 100% of row 1 tubes on all Salem Unit 2 steam generators and any row 2 tubes identified as defective during supplemental testing. The result of plugging these tubes, as well as those non-row 1 tubes identified as defective during the testing, is a net reduction in available RCS flow. As a result, it is anticipated that upon performance of the RCS flow calorimetric following startup from the present refueling outage, RCS flow will fall below, or at best be only marginally above the minimum allowable value. The Limiting Condition for Operation of Specification 3/4.2.3 requires the combination of RCS flow and R1 and R2 be restored to within the allowable operating region of

Figure 3.2-3 in 24 hours or that reactor power be reduced to less than 5% within the next 2 hours.

As can be seen from the above discussion, without the proposed change Salem Unit 2 could be forced into an unnecessary shutdown. Additionally, this condition could not have been reasonably foreseen prior to this time as it is a direct result of work accomplished during the present refueling outage. Further, Salem Unit 2 is scheduled to be synchronized with the grid on October 28, 1988 which is less than 15 days away. It is therefore concluded that the change satisfies the Commission's criteria for consideration as an Emergency License Amendment request.

ENCLOSURE 2

SALEM UNIT NO. 2
SECONDARY SIDE CALORIMETRIC INSTRUMENTATION
PLANT SPECIFIC UNCERTAINTIES FOR COMPARISON WITH WCAP-11579

CALORIMETRIC INSTRUMENTATION UNCERTAINTIES

A. Feed Water Temperature

1. Sensor calibration error $\pm .1^{\circ}F$ = .025% of span {from thermocouple specification}

2. Indicating device error = $\pm 1^{\circ}F$ = .25% of span {from test performed by PSE&G lab}

3. Readability of indicator = $\pm .5^{\circ}F$ = .125% of span {from observation}

4. Temperature effects, Pressure effects, Maintenance and test Equipment error, and rack error = 0

5. Channel Statistical Accuracy (CSA)
   $CSA = [(.025)^2 + (.25 + .125)^2]^{1/2}$
   $= .376\%$ of span

6. Span = $400^{\circ}F$

7. Feed water temperature uncertainty = 400x.376% = $1.5^{\circ}F$

B. Feed Water Flow Differential Pressure

1. Sensor Calibration Accuracy = $[(.5)^2 + (.2)^2]^{1/2}$ = .54% {.5 is from calibration procedure and .2 is from vendor manual for a Rosemount 1151DP6 transmitter.}

2. Sensor Drift = .2% of sensor upper range limit (URL) in 6 months {from vendor manual} or [(.2)x(3000)]/1280 = .469% span.

3. Maintenance and test equipment error = .025% for dead weight tester plus .1% for Digital Volt Meter {from PSE&G Calibration Lab.}

4. Readout device accuracy .08% per calibration procedure plus $\pm$5inches or 5/1280 = .39% readability error

5. Sensor temperature effects = 2.6% per $100^{\circ}F$ or .78% per $30^{\circ}F$

6. Pressure Effects = [.25%/2000psi] x 900psi span = .112%

7. $CSA = [\{.54 + [(.025)^2 + (.1)^2]^{1/2} + .469\}^2 + (.78)^2 + (.08 + .39)^2 + (.025)^2 + (.112)^2]^{1/2}$
   $= 1.44\%$ of span or 1.7% DP

C. Steam Pressure

1. Sensor Calibration Accuracy = $[(.5)^2 + (.25)^2]^{1/2}$ = .56% {.5 is from calibration procedure and .25 is from vendor manual for a Rosemount 1153GD9 transmitter.}

2. Sensor Drift = 25% URL in 6 months {from vendor manual} or (.25)x(3000)/1200 = .625% of span

3. Maintenance and test equipment error = 3psi = 3/1200 = .25% of span {from PSE&G Calibration Lab.}

4. Readout device accuracy 1% per calibration procedure

5. Sensor temperature effects = .75%x(3000) + .5%x(1204)
   = 28.58psi = 2.37% span per 100°F
   = 1.185% per 50°F

6. Loop accuracy = .5% per procedure

7. CSA = $[(.56 + .25 + .625)^2 + (1.185)^2 + (1 + .5)^2]^{1/2}$
   =2.39% of span or 28.7psi

D. Pressurizer Pressure

1. Sensor Calibration Accuracy = $[(.5)^2 + (.25)^2]^{1/2}$ = .56% {.5 is from calibration procedure and .25 is from vendor manual for a Rosemount 1153GD9 transmitter.}

2. Sensor Drift = 25% in 6 months {from vendor manual} or (.25)x(3000)/800 = .94% of span

3. Maintenance and test equipment error = 3psi = 3/800 = .375% of span {from PSE&G Calibration Lab.}

4. Readout device accuracy = (.001)x(20)/4 = .5% {Fluke accuracy}

5. Sensor temperature effects = .75%x(3000) + .5%x(800)
   = 26.5psi = 3.3% span per 100°F
   = 1.66% per 50°F

6. Loop accuracy = .5% per procedure

7. CSA = $[(.56 + .375 + .94)^2 + (1.66)^2 + (.5 + .5)^2]^{1/2}$
   = 2.70% of span or 21.6psi

Comparison of WCAP 11579 Instrument uncertainties with PSE&G Calculated uncertainties:

	FW TEMP ^{o}F	FW DP %	STM PRESS psi	PRZ PRESS psi
INST. SPAN =	400	120	1200	800
WESTINGHOUSE	2.0	1.1	30.1	19.5
PSE&G	1.5	1.7	28.7	21.6

NOTES:

1. All percentage values are percent span unless otherwise noted.

2. 50^{o}F temperature variations were assumed in determining sensor temperature effects for equipment in containment. 30^{o}F temperature variations were assumed for equipment located in the turbine building due to better ventilation.

3. Calorimetrics are assumed to take place within 6 months of sensor calibration. Sensors are calibrated every refueling outage.

4. Feed water pressure is assumed to be 900psi during calorimetric calculations. Errors associated with this assumption have negligible affect upon feed flow and enthalpy. This is because procedures require reactor power to be above 90% during performance of RCS flow measurement. At power levels between 90% and 100% the difference between the assumed feed water pressure and actual feed water temperature will be within the 60 psi uncertainty calculated by Westinghouse. It is also noted that RCS flow measurements are relatively insensitive to feed water pressure.

ENCLOSURE 3

SALEM UNIT NO.2
AFFECTED TECHNICAL SPECIFICATION PAGES

POWER DISTRIBUTION LIMITS

3/4.2.3 RCS FLOW RATE AND R

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R_1, R_2 shall be maintained within the region of allowable operation shown on Figure 3.2-3 for 4 loop operation.

Where:

a. $R_1 = \dfrac{F^N_{\Delta H}}{1.49\ [1.0 + 0.3\ (1.0 - P)]}$,

b. $R_2 = \dfrac{R_1}{[1\text{-}RBP(BU)]}$,

c. $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

d. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate R since Figure 3.2-3 includes measurement uncertainties of ~~3.5%~~ 2.2% for flow and 4% for incore measurement of $F^N_{\Delta H}$.

e. RBP (BU) = Rod Bow Penalty as a function of region average burnup as shown in Figure 3.2-4, where a region is defined as those assemblies with the same loading date (reloads) or enrichment (first core).

APPLICABILITY: MODE 1.

ACTION:

With the combination of RCS total flow rate and R_1, R_2 outside the region of acceptable operation shown on Figure 3.2-3:

a. Within 2 hours:

1. Either restore the combination of RCS total flow rate and R_1, R_2 to within the above limits, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High trip setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

LIMITING CONDITION FOR OPERATION (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparision that the combination of R_1, R_2 and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION items a.2 and/or b., above; subsequent POWER OPERATION may proceed provided that the combination of R_1, R_2 and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation shown on Figure 3.2-3 prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,

2. A nominal 75% of RATED THERMAL POWER, and

3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and R_1, R_2 shall be determined to be within the region of acceptable operation of Figure 3.2-3:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

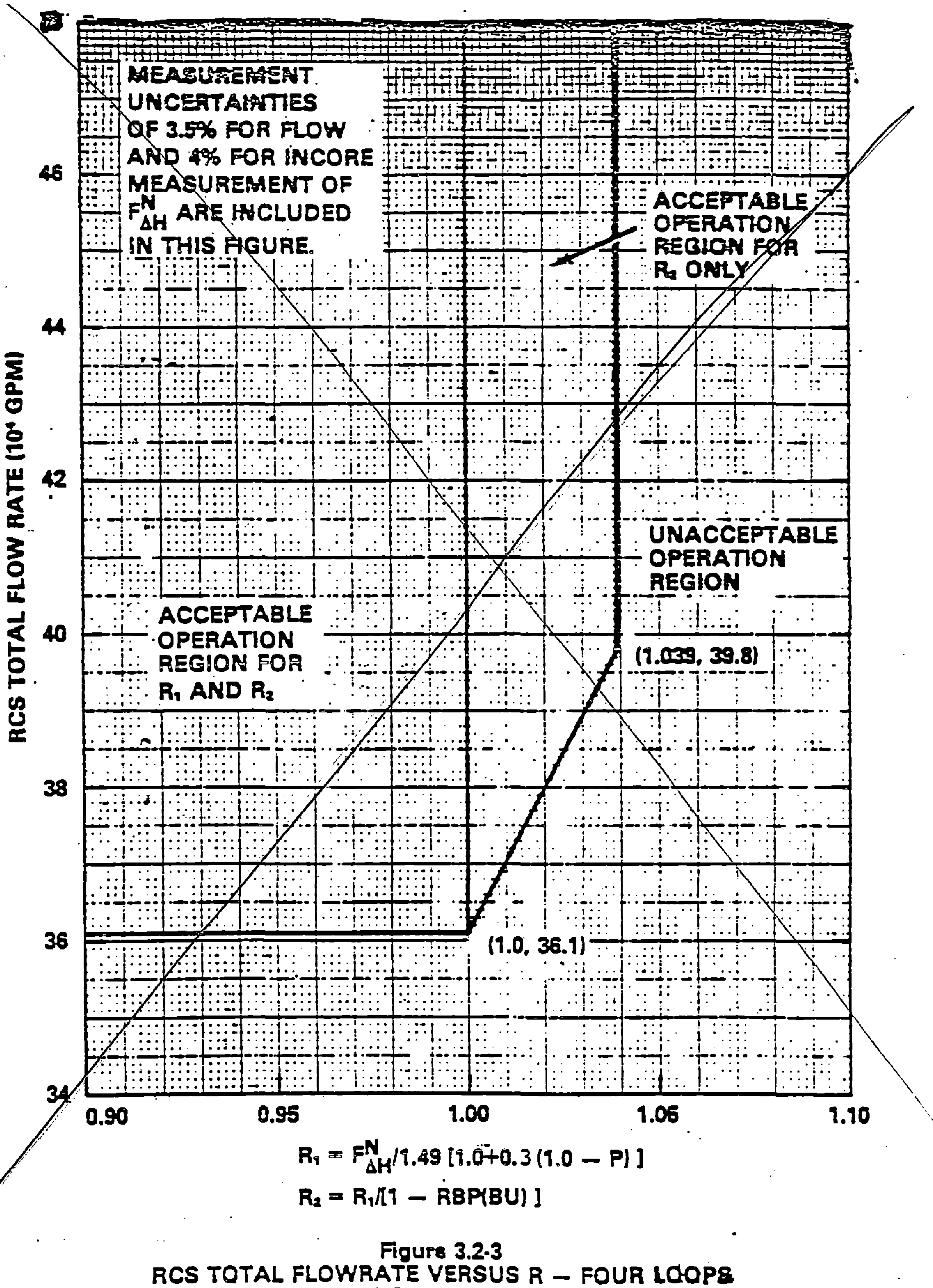
b. At least once per 31 Effective Full Power Days.

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation of Figure 3.2-3 at least once per 12 hours when the values of R_1 and R_2 obtained per Specification 4.2.3.2 are assumed to exist.

4.2.3.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.3.5 The RCS total flow rate shall be determined by measurement at least once per 18 months.

REPLACE WITH NEW FIGURE 3.2-3



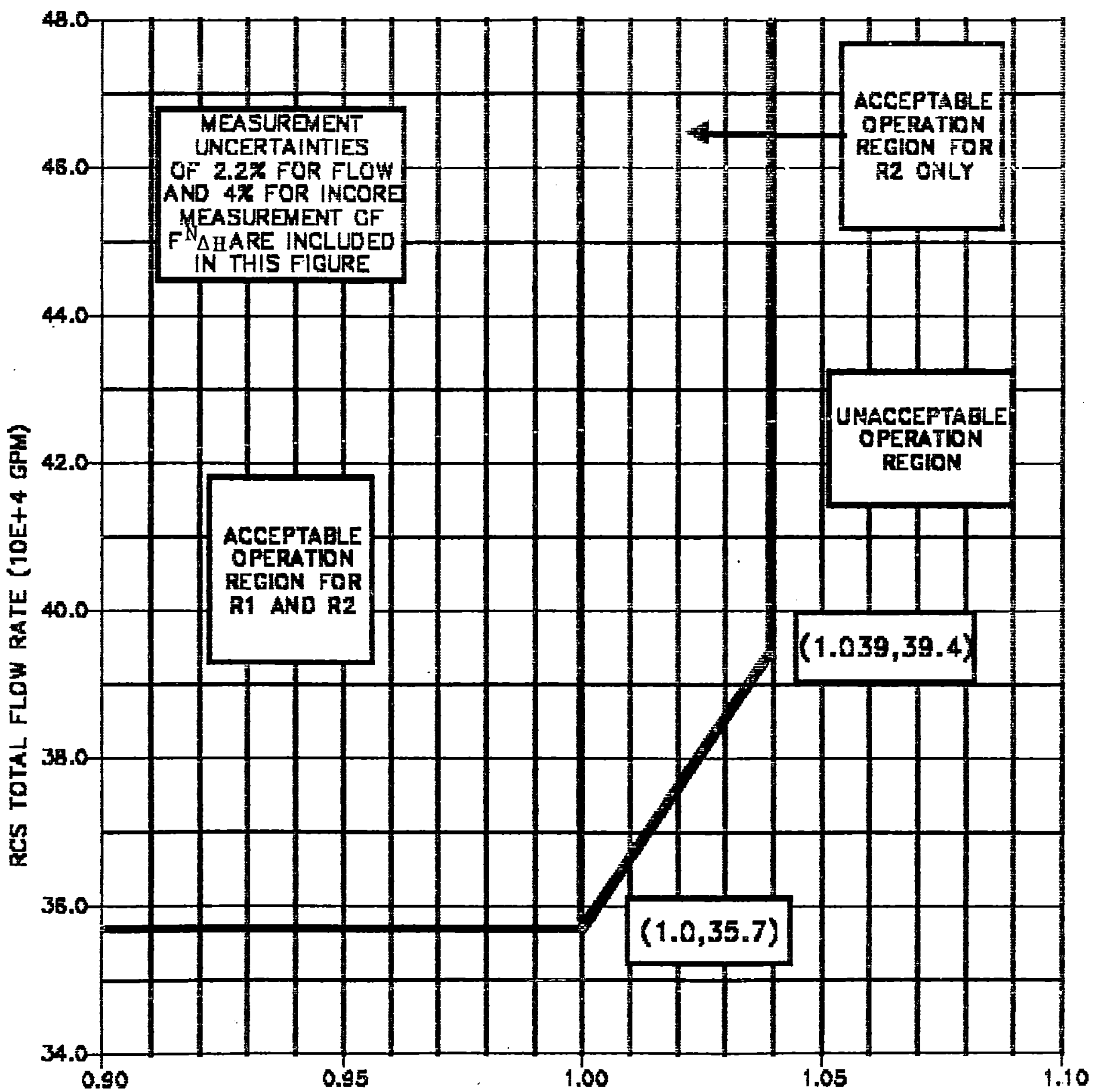
$R_1 = F^N_{\Delta H}/1.49\ [1.0 + 0.3\ (1.0 - P)]$

$R_2 = R_1/[1 - RBP(BU)]$

Figure 3.2-3
RCS TOTAL FLOWRATE VERSUS R – FOUR LOOPS IN OPERATION

FIGURE 3.2-3

RCS TOTAL FLOWRATE VERSUS R–FOUR LOOPS IN OPERATION



$R1 = F^N_{\Delta H}/1.49[1.0+0.3(1.0 - P)]$

$R2 = R1 /[1 - RBP(BU)]$

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR $F_Q(Z)$, RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux and nuclear enthalpy hot channel factors and RCS flow rate ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rod in a single group move together with no individual rod insertion differing by more than ± 12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.5.

c. The control rod insertion limits of Specifications 3.1.3.4 and 3.1.3.5 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided conditions a. through d., above, are maintained. As noted in Figure 3.2-3, RCS flow rate and $F^N_{\Delta H}$ may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the measured $F^N_{\Delta H}$ is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R_1, as calculated in 3.2.3 and used in Figure 3.2-3, accounts for $F^N_{\Delta H}$ less than or equal to 1.49. This value is used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed. R_2, as defined, allows for the inclusion of a penalty for rod bow on DNBR only, thus knowing the "as measured" values of $F^N_{\Delta H}$ and RCS flow allows for "tradeoffs" in excess of R equal to 1.0 for the purpose of offsetting the rod bow DNBR penalty.

Fuel rod bowing reduces the value of DNB ratio. Sufficient credit is available to offset this reduction. This credit comes from generic design margins totaling 9.1% and 3% margin in the difference between the 1.3 DNBR safety limit and the minimum DNBR calculated for the Complete Loss of Flow event. The penalties applied to $F^N_{\Delta H}$ to account for Rod Bow (Figure 3.2-4) as a function of burnup are consistent with those described in Mr. John F. Stolz's (NRC) letter to T. M. Anderson (Westinghouse) dated April 5, 1979 and W 8691 Rev. 1 (partial rod bow test data).

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance. When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of Figure 3.2-3. ~~Measurement errors of 3.5% for RCS total flow rate and 4% for $F_{\Delta H}$ have been allowed for in determination of the design DNBR value.~~ INSERT 1

The 12 hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation shown in Figure 3.2-3.

The radial peaking factor $F_{xy}(z)$ is measured periodically to provide assurance that the hot channel factor, FQ(z), remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}), as provided in the Radial Peaking Factor Limit Report per specification 6.9.1.10, was determined from expected power control maneuvers over the full range of burnup conditions in the core.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

INSERT 1

Figure 3.2-3 is adjusted to allow for a 2.2% error in RCS flow rate and a 4% error in for $F^N_{\Delta H}$. The correction for RCS flow was reduced as a result of the replacement of the RTD Bypass Loops with thermowell RTDs. The detailed justification for the reduction is contained in WCAP-11579 (Proprietary) dated September 1987.