PSE&G

Ref. LCR 79-06

Public Service Electric and Gas Company 80 Park Place Newark, N.J. 07101 Phone 201/430-7000

March 2, 1979

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Mr. Albert Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Gentlemen:

REGULATORY DOCKET FILE COPY

CYCLE 2 RELOAD ANALYSIS
FACILITY OPERATING LICENSE DPR-70
UNIT NO. 1
SALEM GENERATING STATION
DOCKET NO. 50-272

Salem Unit No. 1 is currently in its first cycle of operation with a refueling outage scheduled to commence on March 31, 1979. Cycle 1 operation will be terminated within a cycle burnup range of 14,400 to 15,600 MWD/MTU. Startup of cycle 2 is expected to occur in mid June 1979. This letter is to advise you of PSE&G's review of and plans regarding the Salem No. 1, cycle 2 reload core.

The cycle 2 reload core will consist of forty (40) new Westinghouse 17 x 17 fuel assemblies. Two (2) of these will be of the optimized fuel assembly design and will be inserted as part of Westinghouse's "Optimized Fuel Assembly Demonstration Program" (WCAP-9286).

The Salem No. 1, cycle 2 reload core was designed such that those incidents analyzed and reported in the Salem FSAR which could potentially be affected by the fuel reload have been reviewed for the cycle 2 design. This review was performed in accordance with the Westinghouse reload methodology as outlined in the March 1978 Westinghouse topical report entitled "Westinghouse Reload Safety Evaluation Methodology" (WCAP-9272). The small break LOCA is presently being reanalyzed to confirm the ECCS analysis for cycle 2. The results of this reanalysis will be submitted to you after it is completed, which is anticipated to be during the latter part of March 1979. PSE&G has reviewed in detail the bases of the reload analysis and the Westinghouse Reload Safety Evaluation (RSE) Report with Westinghouse. The review of all incidents completed to date has demonstrated and the small break LOCA reanalysis is expected to demonstrate that the results of all the postulated events are within allowable limits.

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The reload core design will be verified during the startup physics testing program. This program will include, but not be limited to, the following tests:

1. Control rod drive tests and drop time
2. Critical boron concentration measurements
3. Control rod bank worth measurements
4. Moderator temperature coefficient measurement
5. Power coefficient measurement, and
6. Startup power distribution measurements using the incore flux mapping system.

In accordance with the Atomic Energy Act of 1954, as amended and the regulations thereunder, we hereby transmit copies of our request for amendment and our analysis of the changes to Facility Operating License DPR-70 for Salem Generating Station Unit No. 1.

This request consists of proposed changes to the Safety Technical Specifications (Appendix A), pertaining to the following areas: Axial Flux Difference, Heat Flux Hot Channel Factor - $F_Q(Z)$, Nuclear Enthalpy Hot Channel Factor - $F^N_{\Delta H}$, and Reactor Core.

This change request package is deemed to involve several Class III changes (each involving a single safety issue and deemed not to involve a significant hazards consideration) and, therefore, is determined to be a Class IV amendment as defined by 10CFR 170.22. A check in the amount of $12,300 is enclosed.

This submittal includes three signed originals and 40 copies.

Very truly yours,

Frank P. Librizzi
General Manager -
Electric Production

Ref. LCR 79-06

U.S. NUCLEAR REGULATORY COMMISSION
DOCKET NO. 50-272

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
FACILITY OPERATING LICENSE NO. DPR-70
NO. 1 UNIT
SALEM GENERATING STATION

Public Service Electric and Gas Company hereby submits proposed changes to Facility Operating License No. DPR-70 for Salem Generating Station, Unit No. 1. This change request relates to Safety Technical Specifications (Appendix A) of the Operating License, and pertains to changes required for cycle 2 operation.

Respectfully submitted,

PUBLIC SERVICE ELECTRIC AND GAS COMPANY

BY: Frederick W. Schneider
FREDERICK W. SCHNEIDER
VICE PRESIDENT

Ref. LCR 79-06

STATE OF NEW JERSEY)
) SS:
COUNTY OF ESSEX )

FREDERICK W. SCHNEIDER, being duly sworn according to law deposes and says:

I am a Vice President of Public Service Electric and Gas Company, and as such, I signed the request for change to FACILITY OPERATING LICENSE NO. DPR-70.

The matters set forth in said change request are true to the best of my knowledge, information, and belief.

Frederick W Schneider
FREDERICK W. SCHNEIDER

Subscribed and sworn to before me
this 5th day of March, 1979.

Barbara Vallee
Notary Public of New Jersey

My commission expires on Nov 8, 1983

BARBARA VALLEE
A NOTARY PUBLIC OF NEW JERSEY
My Commission Expires [illegible] 83

PROPOSED CHANGE
AXIAL FLUX DIFFERENCE
TECHNICAL SPECIFICATION
SALEM UNIT NO. 1

## Description of Change

During the first 2700 MWD/MTU of cycle 2 operation, the indicated axial flux difference will be restricted to less than +7.5% at rated thermal power. This allowable axial flux difference will increase by 1.0% for each 1.0% reduction in power level.

## Reason for Change

This restriction is necessary to ensure that the core peaking factor limits are met during cycle 2 operation.

## Safety Evaluation

This change is required to ensure that the peaking factor limits assumed in the bases of the technical specifications and in the FSAR are met during cycle 2. Therefore, this change does not involve an unreviewed safety question.

## AXIAL FLUX DIFFERENCE (AFD)

### LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE shall be maintained within a +5% target band (flux difference units) about the target flux difference.

Add attached addition to page 3/4 2-1

APPLICABILITY: MODE 1 ABOVE 50% RATED THERMAL POWER*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the limits of ~~+5% target band about the target flux difference~~ 3.2.1 above and with THERMAL POWER:

1. Above 90% of RATED THERMAL POWER, within 15 minutes:

   a) Either restore the indicated AFD to within the target band limits, or

   b) Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

2. Between 50% and 90% of RATED THERMAL POWER:

   a) POWER OPERATION may continue provided:

      1) The indicated AFD has not been outside of the limits of ~~+5% target band~~ 3.2.1 above for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

      2) The indicated AFD is within the limits shown on Figure 3.2-1. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

   b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limits of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

*See Special Test Exception 3.10.2

## LIMITING CONDITION FOR OPERATION (Continued)

b. THERMAL POWER shall not be increased above 90% of RATED THERMAL POWER unless the indicated AFD is within the ~~± 5% target band~~ Limits of 3.2.1 above and ACTION 2.a) 1), above has been satisfied.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the Limits of 3.2.1 above ~~± 5% target band~~ for more than 1 hour penalty deviation cumulative during the previous 24 hours.

## SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

   1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

   2. At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its limits of 3.2.1 above ~~± 5% target band~~ when at least 2 of 4 or 2 of 3 OPERABLE excore channels are indicating the AFD to be outside the limits of 3.2.1 above ~~target band~~. Penalty deviation outside of the ~~± 5% target band~~ limits of 3.2.1 above shall be accumulated on a time basis of:

a. One minute penalty deviation for each one minute of POWER OPERATION outside of the limits of 3.2.1 above ~~target band~~ at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each one minute of POWER OPERATION outside of the limits of 3.2.1 above ~~target band~~ at THERMAL POWER levels bel 50% of RATED THERMAL POWER.

Proposed addition to page 3/4 2-1

In addition the following restriction will be required during the first 72 EFPD (2700 MWD/MTU) operation of cycle 2. The indicated axial flux difference will be maintained less than +7.5% at RATED THERMAL POWER with the allowed axial flux difference increasing by 1.0% for each 1.0% reduction in power level.

PROPOSED CHANGES
HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$
TECHNICAL SPECIFICATIONS
SALEM UNIT NO. 1

Description of Changes

1. Revise the third line segment of the K(Z) curve (figure 3.2-2).

2. Revise the $F_{xy}$ limits as contained in this section of the technical specifications.

Reason for Changes

1. This revision is necessary due to violation of the $F_Q(Z)$ limits during cycle 2 operation with the current third line segment of the K(Z) curve.

2. Recent analyses for Westinghouse plants which undertake reload cycle operations with an $F_{xy}$ technical specification show the need to revise these limits upward for other than cycle 1 operation. Salem No. 1 is one of the first Westinghouse units with an $F_{xy}$ limit to reload and a revision in this limit is required to avoid exceeding $F_{xy}$ limits during the second and subsequent cycles.

Safety Evaluation

1. The revised K(Z) curve has been used in the design of cycle 2. Analysis of cycle 2 is expected to verify the validity of the revised K(Z) curve for cycle 2 operation. Because of this change the small break LOCA analysis must be redone. This analysis, using currently approved Westinghouse analysis models will be completed in March 1979, and is expected to confirm the validity of the revised K(Z) curve. The results of this reanalysis will be submitted after it is completed. If the results are as anticipated the LOCA analysis for cycle 2 will be confirmed using the revised K(Z) curve and therefore, this change will not constitute an unreviewed safety question. The revised third line segment will be reverified for each reload beyond cycle 2.

2. $F_Q(Z)$, which is the primary power distribution parameter in the Technical Specifications for LOCA protection, is deter-

mined by the product of the radial $F_{xy}$ and the axial F(Z) peaking factors. In the event the $F_{xy}$ is exceeded, continued operation is allowed provided the $F_Q(Z)$ limit is met. Since reload cores exhibit flatter axial shapes, as indicated by lower F(Z), the revised $F_{xy}$ limits will still result in $F_Q(Z)$ being within allowable limits.

For operation of cycle 2 and subsequent cycles of the Salem No. 1 core, bounding values of the peaking factor $F_Q(Z)$ x (relative power) were calculated as a function of elevation by assuming various load follow transients on the reactor through insertion and withdrawal of control rod Banks C and D. The effects of the accompanying variation in axial xenon and power distributions were also considered as described in the Salem FSAR. Both beginning and end of cycle conditions were included in the cycle 2 calculations, and several different histories of operation were assumed in calculating effects of load follow transients on the axial power distribution. Results of these calculations demonstrate that the $F_Q(Z)$ limits will not be exceeded during operation of cycle 2. Further evaluations indicate that the proposed $F_{xy}$ change will not cause the $F_Q(Z)$ limits to be exceeded in projected reload cycles beyond cycle 2. The $F_Q(Z)$ limit envelope will be reverified for each reload beyond cycle 2 to confirm this projection. Therefore, this change does not involve an unreviewed safety question.

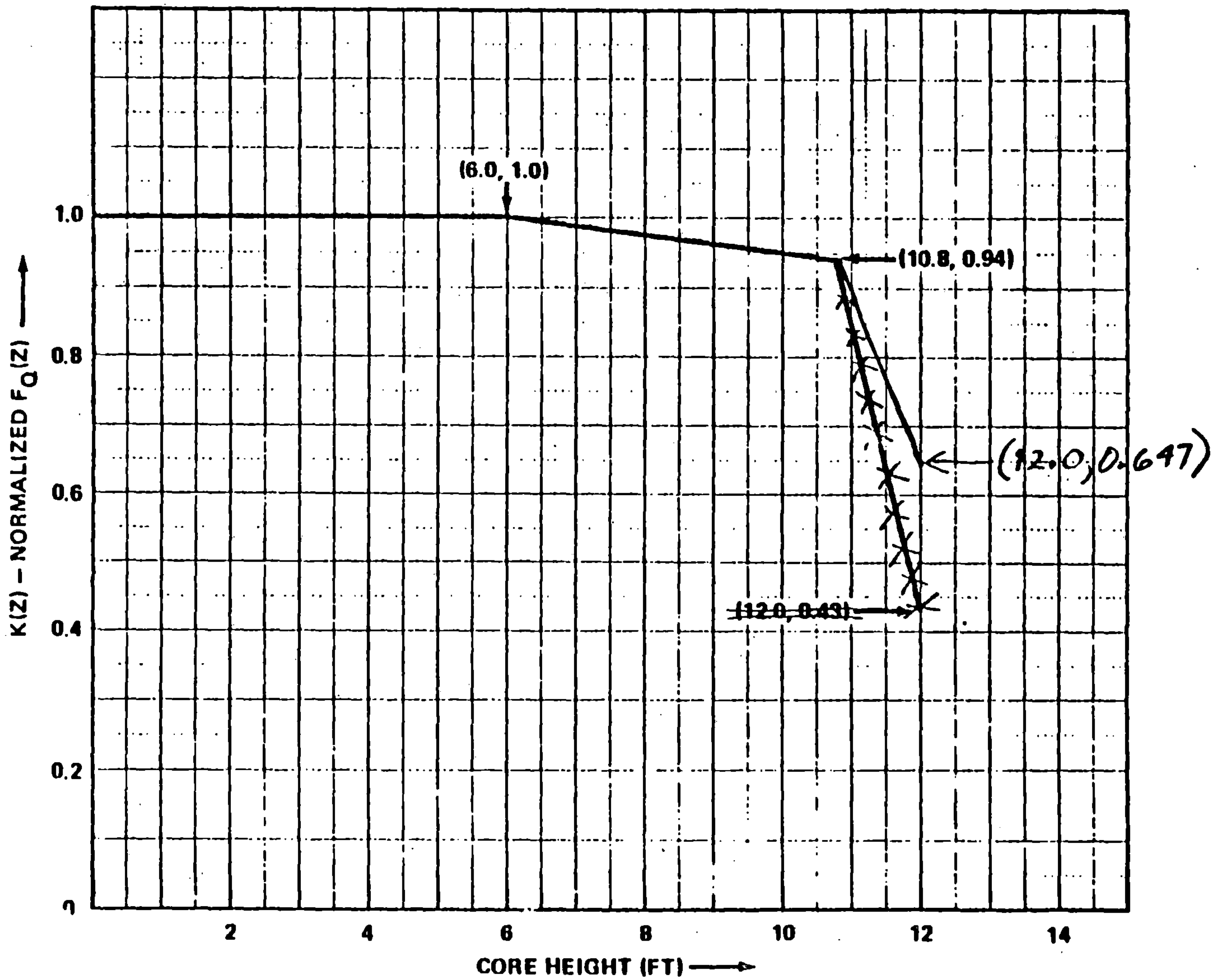


**FIGURE 3.2-2**

**K(Z) – NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT**

b) At least once per 31 EFPD, whichever occurs first.

2. When the $F_{xy}^{C}$ is less than or equal to the $F_{xy}^{RTP}$ limit for the appropriate measured core plane, additional power distribution maps shall be taken and $F_{xy}^{C}$ compared to $F_{xy}^{RTP}$ and $F_{xy}^{L}$ at least once per 31 EFPD.

e. The $F_{xy}$ limits for RATED THERMAL POWER within specific core planes shall be:

~~1. $F_{xy}^{RTP} \leq 1.71$ for all core planes containing either bank "D" control rods or any part length rods, and~~

*Add attached addition to Page 3/4 2-7*

2. $F_{xy}^{RTP} \leq$ ~~1.55~~ *1.65* for all unrodded core planes.

f. The $F_{xy}$ limits of e, above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15%, inclusive.
2. Upper core region from 85 to 100% inclusive.
3. Grid plane regions at 17.8 $\pm$ 2%, 32.1 $\pm$ 2%, 46.4 $\pm$ 2%, 60.6 $\pm$ 2% and 74.9 $\pm$ 2%, inclusive.
4. Core plane regions within $\pm$ 2% of core height ($\pm$ 2.88 inches) about the bank demand position of the bank "D" or part length control rods.

g. Evaluating the effects of $F_{xy}$ on $F_Q(Z)$ to determine if $F_Q(Z)$ is within its limit whenever $F_{xy}^{C}$ exceeds $F_{xy}^{L}$.

4.2.2.3 When $F_Q(Z)$ is measured pursuant to specification 4.10.2.2, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

Proposed addition to page 3/4 2-7

1. For all core planes containing bank "D" control rods:

   a) $F_{xy}^{RTP} \leq 1.92$ for core elevations up to 2.0 ft.

   b) $F_{xy}^{RTP} \leq 1.89$ for core elevations from 2.0 to 12.0 ft., and

## 3/4.2 POWER DISTRIBUTION LIMITS

## BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core $\geq$ 1.30 during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of hot channel factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

*Add attached addition to page B 3/4 2-1*

### 3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope of 2.32 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions with the part length control rods withdrawn from the core. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

### 3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL ~~FACTORS~~ AND RADIAL ~~$F_Q(Z)$ and $F^N_{\Delta H}$~~ PEAKING FACTORS - $F_Q(Z)$, $F^N_{\Delta H}$ and $F_{xy}(Z)$

The limits on heat flux and nuclear enthalpy hot channel factors 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 hot channel factors are 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 hot channel factor limits are maintained provided:

a. Control rod in a single group move together with no individual rod insertion differing by more than $\pm$ 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 3.1.3.5 and 3.1.3.6 are maintained.

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

The relaxation in $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits provided conditions a thru d above, are maintained.

When an $F_Q$ measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

When $F^N_{\Delta H}$ is measured, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system. The specified limit for $F^N_{\Delta H}$ also contains an 8% allowance for uncertainties which mean that normal operation will result in $F^N_{\Delta H} \leq 1.55/1.08$. The 8% allowance is based on the following considerations:

a. abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than $F_Q$,

b. although rod movement has a direct influence upon limiting $F_Q$ to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup physics tests can be compensated for in $F_Q$ by restricting axial flux distributions. This compensation for $F^N_{\Delta H}$ is less readily available.

Add attached addition to page B 3/4 2-5

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.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in $F_Q$ is depleted. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on $F_Q$ is reinstated by reducing the power by 3 percent from RATED THERMAL POWER for each percent of tilt in excess of 1.0.

Proposed addition to page B 3/4 2-1

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

Proposed addition to page B 3/4 2-5

The radial peaking factor, $F_{xy}(Z)$, is measured periodically to provide additional assurance that the hot channel factor, $F_Q(Z)$, remains within its limit. The $F_{xy}(Z)$ limits were determined from expected power control maneuvers over the full range of burnup conditions in the core.

PROPOSED CHANGE
NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$
TECHNICAL SPECIFICATIONS
SALEM UNIT NO. 1

Description of Change

Revise the $F^N_{\Delta H}$ technical specification, incorporating the rod bow penalty curve that is currently approved by the NRC. This curve has been documented as having generic application for Westinghouse 17 x 17 plants in a Westinghouse submittal, NS-TMA-1760, "Fuel Rod Bowing", T. M. Anderson (Westinghouse) to J. F. Stolz (NRC), dated April 19, 1978. This change would result in a reduction of the current rod bow penalty applied to Salem No. 1.

Reason for Change

To reduce the rod bow penalty (especially for second cycle fuel) to accommodate expected $F^N_{\Delta H}$ values that will be encountered at rated thermal power during cycle 2 operation.

Safety Evaluation

This change only incorporates recent information on rod bow penalties and does not reduce the margin of safety as defined in the basis for this specification or as defined in the FSAR. Therefore, this change does not constitute an unreviewed safety question.

## NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

### LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq 1.55\,[1.0 + 0.2\,(1-P)]\,[1 - RBP]$$

where $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

RBP = Rod Bow Penalty as shown in figure 3.2-3

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate thru in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a. or b. above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL power and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

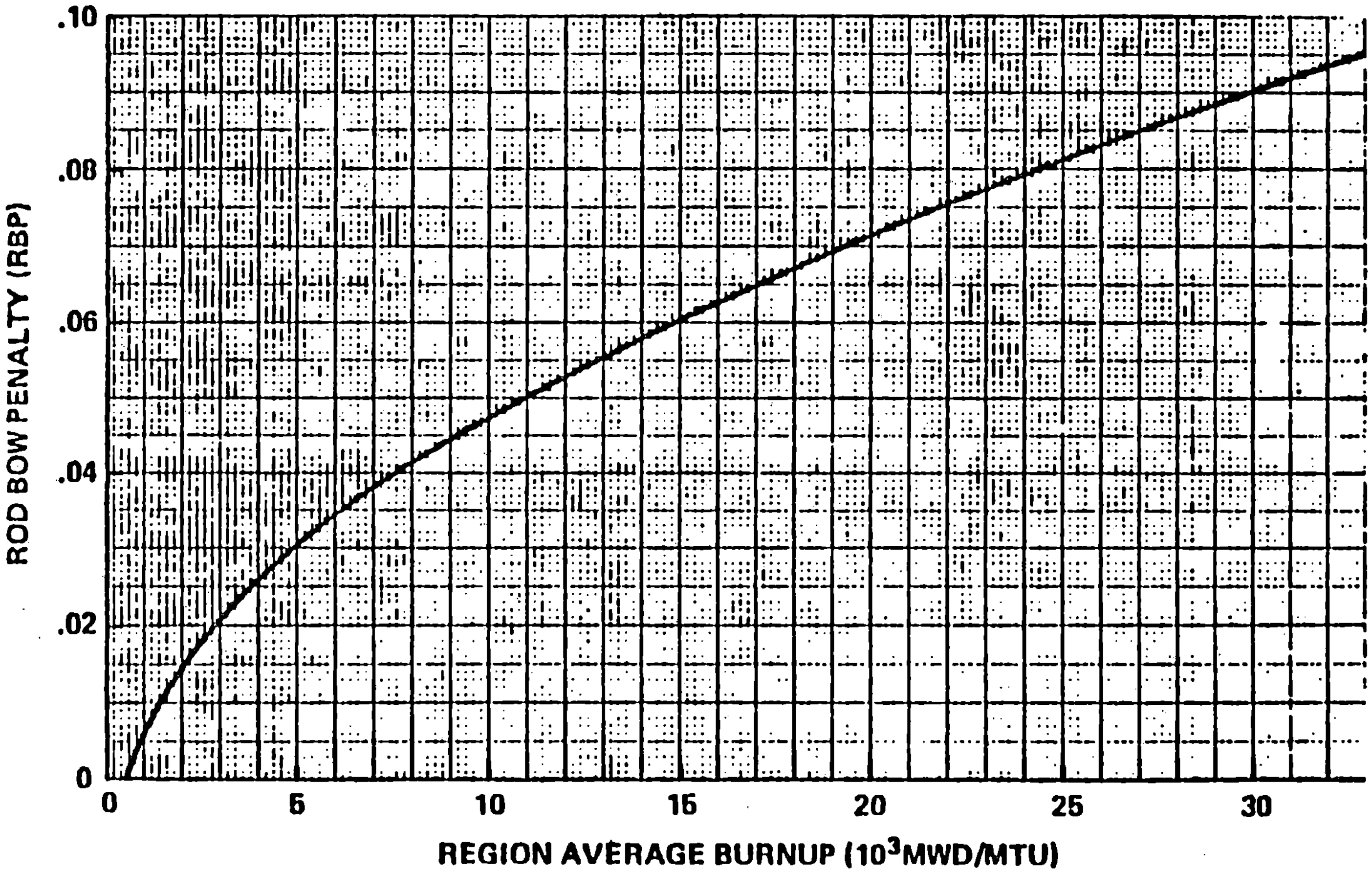


FIGURE 3.2-~~X~~ 3

ROD BOW PENALTY AS A FUNCTION OF BURNUP

PROPOSED CHANGE
REACTOR CORE
TECHNICAL SPECIFICATION
SALEM UNIT NO. 1

## Description of Change

The current technical specifications state that each fuel rod shall contain a "nominal total weight of 1743 grams of uranium". This should be revised to read a "maximum total weight of 1766 grams of uranium". This change is consistent with the wording contained in the June 15, 1978 version of the Westinghouse Standard Technical Specifications (STS).

## Reason for Change

PSE&G is going to be inserting two demonstration assemblies in Salem No. 1 during cycle 2. These assemblies are of the new Westinghouse fuel assembly design known as the optimized fuel assembly. These optimized fuel assemblies have a nominal total weight of uranium of about eight percent less than the standard Westinghouse 17 x 17 fuel assembly.

## Safety Evaluation

This proposed change will only allow for a reduction of the weight of uranium, while maintaining the same limits on enrichment and cannot adversely affect the safety of the unit. Therefore, this change does not involve an unreviewed safety question.

This wording change also has the generic approval of the NRC in the Westinghouse STS.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 47 psig and an air temperature of 271°F.

## 5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy -4. Each fuel rod shall have a nominal active fuel length of 143.7 inches and maximum contain a ~~nominal~~ total weight of ~~1743~~ 1766 grams uranium. The initial core loading shall have a maximum enrichment of 3.35 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.5 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 53 full length and 8 part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The part length control rod assemblies shall contain a nominal 36 inches of absorber material at their lower ends. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing. The balance of the void length in the part length rods shall contain aluminum oxide.

## 5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained: