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Technical Specification 6.9.1.9

LR-N11-0331

U.S. Nuclear Regulatory Commission
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Washington, D.C. 20555-001

Salem Nuclear Generating Station Unit 1
Facility Operating License No. DPR-70
NRC Docket No. 50-272

Subject: Core Operating Limits Report - Cycle 22

In accordance with section 6.9.1.9 of the Salem Unit 1 Technical Specifications, PSEG Nuclear LLC submits the Core Operating Limits Report (COLR) for Salem Unit 1 Cycle 22 in Attachment 1 to this letter.

There are no commitments contained in this letter. Should you have any questions regarding this submittal, please contact Mr. B. Thomas at (856) 339-2022.

Sincerely,

A handwritten signature in black ink, appearing to read "C. Fricker", is written over a faint, larger signature.

Carl J. Fricker
Site Vice President - Salem

Attachments (1)

ADD
NRC

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cc Mr. W. Dean, Administrator - Region I
 Mr. R. Ennis, Licensing Project Manager - Salem
 USNRC Senior Resident Inspector - Salem (X24)
 Mr. P. Mulligan, Manager IV
 Mr. H. Berrick, Salem Commitment Tracking Coordinator
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Bcc: President and Chief Nuclear Officer
Vice President Operations Support
Site Vice President – Salem
Director - Nuclear Oversight
Director - Regulatory Affairs
Director - Nuclear Fuels
Plant Manager - Salem
Regulatory Assurance Manager - Salem
Licensing Manager

Attachment 1
LR-N11-0331

Core Operating Limits Report For Salem Unit 1, Cycle 22

**COLR SALEM 1
Revision 3
August 2011**

Core Operating Limits Report for Salem Unit 1, Cycle 22

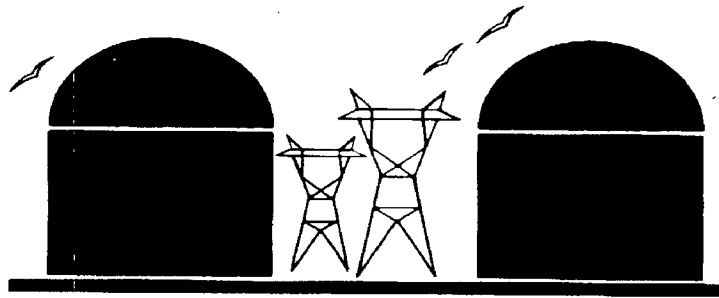


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1.0 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for Salem Unit 1 Cycle 22 has been prepared in accordance with the requirements of Technical Specification 6.9.1.9.

The Technical Specifications affected by this report are listed below:

- 3.1.1.4 Moderator Temperature Coefficient
- 3.1.3.5 Control Rod Insertion Limits
- 3.2.1 Axial Flux Difference
- 3.2.2 Heat Flux Hot Channel Factor - $F_Q(Z)$
- 3.2.3 Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}^N$
- 3.9.1 Boron Concentration

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. These limits have been developed using the NRC-approved methodologies specified in Technical Specification 6.9.1.9 and in Section 3.0 of this report.

2.1 Moderator Temperature Coefficient (Specification 3.1.1.4)

2.1.1 The Moderator Temperature Coefficient (MTC) limits are:

The BOL/ARO/HZP-MTC shall be less positive than or equal to $0 \Delta k/k/^\circ F$.

The EOL/ARO/RTP-MTC shall be less negative than or equal to $-4.3 \times 10^{-4} \Delta k/k/^\circ F$.

2.1.2 The MTC Surveillance limit is:

The 300 ppm/ARO/RTP-MTC should be less negative than or equal to $-3.6 \times 10^{-4} \Delta k/k/^\circ F$.

where: BOL stands for Beginning of Cycle Life

ARO stands for All Rods Out

HZP stands for Hot Zero THERMAL POWER

EOL stands for End of Cycle Life

RTP stands for Rated THERMAL POWER

2.2 Control Rod Insertion Limits (Specification 3.1.3.5)

2.2.1 The control rod banks shall be limited in physical insertion as shown in Figure 1.

2.3 Axial Flux Difference (Specification 3.2.1)

[Constant Axial Offset Control (CAOC) Methodology]

2.3.1 The Axial Flux Difference (AFD) target band shall be (+6%, -9%).

2.3.2 The AFD Acceptable Operation Limits are provided in Figure 2.

2.4 Heat Flux Hot Channel Factor - $F_Q(Z)$ (Specification 3.2.2)

[F_{xy} Methodology]

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{P} * K(Z) \text{ for } P > 0.5$$

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{0.5} * K(Z) \text{ for } P \leq 0.5$$

where: $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

2.4.1 $F_Q^{RTP} = 2.40$

2.4.2 $K(Z)$ is provided in Figure 3.

2.4.3 $F_{xy}^L = F_{xy}^{RTP} [1.0 + PF_{xy}(1.0 - P)]$

where: $F_{xy}^{RTP} = 1.98$	for unrodded upper core planes 1 through 6
1.77	for unrodded upper core planes 7 through 31
1.80	for unrodded lower core planes 32 through 55
1.98	for unrodded lower core planes 56 through 61
2.07	for the core planes containing Bank D control rods

$PF_{xy} = 0.3$

- 2.4.4 If the Power Distribution Monitoring System (PDMS) is used for core power distribution surveillance and is OPERABLE, as defined in Technical Specification 3.3.3.14, the uncertainty, U_{FQ} , to be applied to the Heat Flux Hot Channel Factor $F_Q(z)$ shall be calculated by the following formula:

$$U_{FQ} = \left(1.0 + \frac{U_Q}{100.0} \right) \cdot U_e$$

where:

U_Q = Uncertainty for power peaking factor as defined in equation 5-19 of Analytical Method 3.5.

U_e = Engineering uncertainty factor.
= 1.03

Note: U_{FQ} = PDMS Surveillance Report Core Monitor Fxy Uncertainty in %.

- 2.4.5 If the INCORE movable detectors are used for core power distribution surveillance, the uncertainty, U_{FQ} , to be applied to the Heat Flux Hot Channel Factor $F_Q(z)$ shall be calculated by the following formula:

$$U_{FQ} = U_{qu} \cdot U_e$$

where:

U_{qu} = Base F_Q measurement uncertainty.
= 1.05

U_e = Engineering uncertainty factor.
= 1.03

2.5 Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}^N$ (Specification 3.2.3)

$$F_{\Delta H}^N = F_{\Delta H}^{RTP} [1.0 + PF_{\Delta H} (1.0 - P)]$$

where: $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

2.5.1 $F_{\Delta H}^{RTP} = 1.65$

2.5.2 $PF_{\Delta H} = 0.3$

2.5.3 If the Power Distribution Monitoring System (PDMS) is used for core power distribution surveillance and is OPERABLE, as defined in Technical Specification 3.3.3.14, the uncertainty, $U_{F_{\Delta H}}$, to be applied to the Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^N$, using the PDMS shall be the greater of 1.04 or as calculated by the following formula:

$$U_{F_{\Delta H}} = 1.0 + \frac{U_{\Delta H}}{100.0}$$

where: $U_{\Delta H}$ = Uncertainty for enthalpy rise hot channel factor as defined in equation 5-19 of Analytical Method 3.5.

2.5.4 If the INCORE movable detectors are used for core power distribution surveillance, the uncertainty, $U_{F_{\Delta H}}$, to be applied to the Nuclear Enthalpy Rise Hot Channel Factor $F_{\Delta H}^N$ shall be calculated by the following formula:

$$U_{F_{\Delta H}} = U_{F_{\Delta Hm}}$$

where:

$$U_{F_{\Delta Hm}} = \text{Base } F_{\Delta H} \text{ measurement uncertainty.} \\ = 1.04$$

2.6 Boron Concentration (Specification 3.9.1)

A Mode 6 boron concentration, maintained at or above 2160 ppm, in the Reactor Coolant System, the fuel storage pool, the refueling canal, and the refueling cavity ensures the most restrictive of the following reactivity conditions is met:

- a) A K-effective (K_{eff}) of 0.95 or less at All Rods In (ARI), Cold Zero Power (CZP) conditions with a 1% $\Delta k/k$ uncertainty added.
- b) A K_{eff} of 0.99 or less at All Rods Out (ARO), CZP conditions with a 1% $\Delta k/k$ uncertainty added.
- c) A boron concentration of greater than or equal to 2000 ppm, which includes a 50 ppm conservative allowance for uncertainties.

3.0 ANALYTICAL METHODS

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents.

3.1 WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, July 1985 (W proprietary), Methodology for Specifications listed in 6.9.1.9.a. Approved by Safety Evaluation dated May 28, 1985.

3.2 WCAP-8385, Power Distribution Control and Load Following Procedures – Topical Report, September 1974 (W proprietary), Methodology for Specification 3/4.2.1 Axial Flux Difference. Approved by Safety Evaluation dated January 31, 1978.

3.3 WCAP-10054-P-A, Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code, August 1985 (W proprietary), Methodology for Specification 3/4.2.2 Heat Flux Hot Channel Factor. Approved for Salem by NRC letter dated August 25, 1993.

- 3.4 WCAP-10266-P-A, Revision 2, The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code, March 1987 (W proprietary), Methodology for Specification 3/4.2.2 Heat Flux Hot Channel Factor. Approved by Safety Evaluation dated November 13, 1986.
- 3.5 WCAP-12472-P-A, BEACON - Core Monitoring and Operations Support System, August 1994 (W proprietary). Approved by Safety Evaluation dated February 16, 1994.
- 3.6 CENPD-397-P-A, Revision 01, Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology, May 2000. Approved by Safety Evaluation dated March 20, 2000.

4.0 REFERENCES

1. Salem Nuclear Generating Station Unit No. 1, Amendment No. 297, License No. DPR-70, Docket No. 50-272.

FIGURE 1

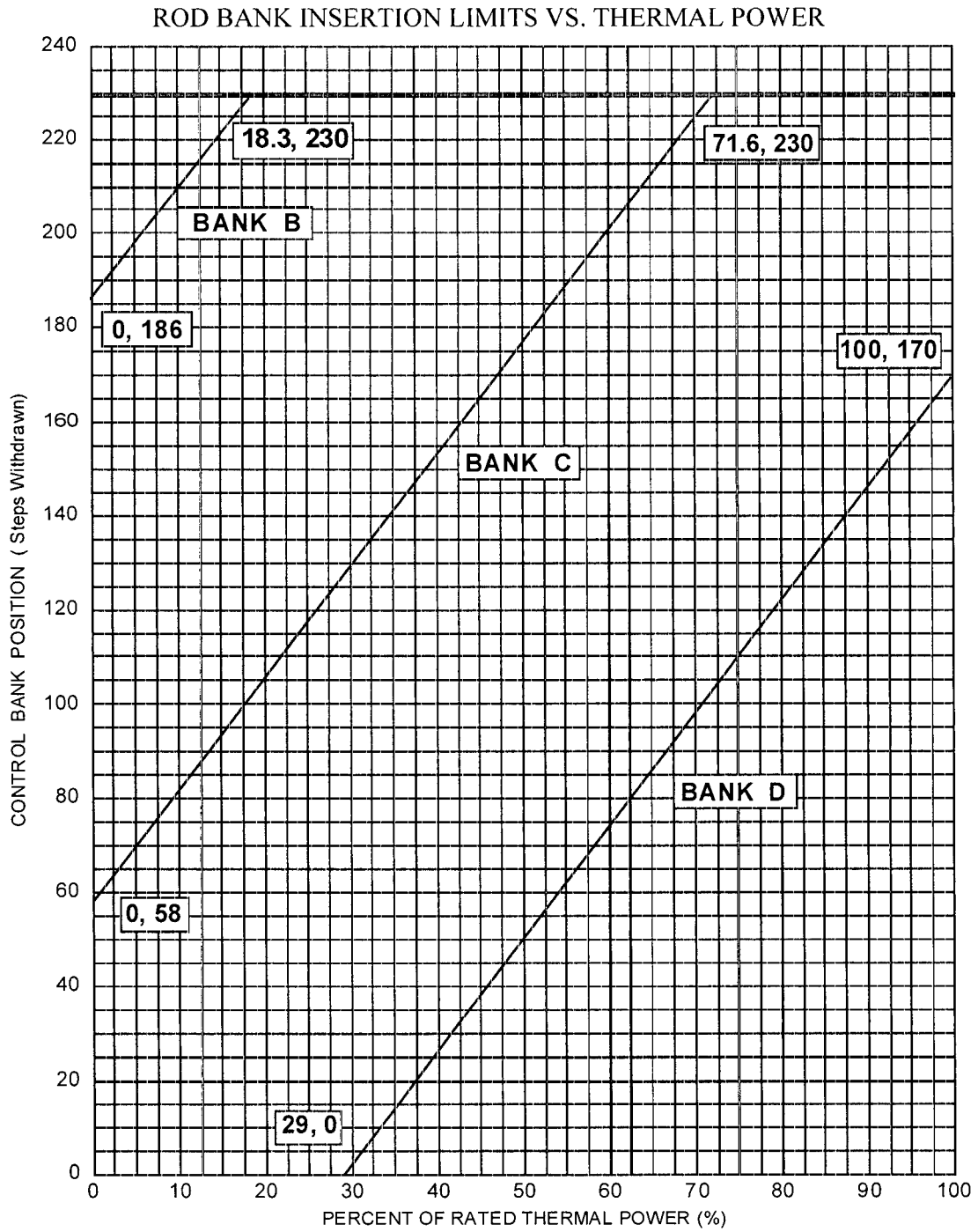


FIGURE 2

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

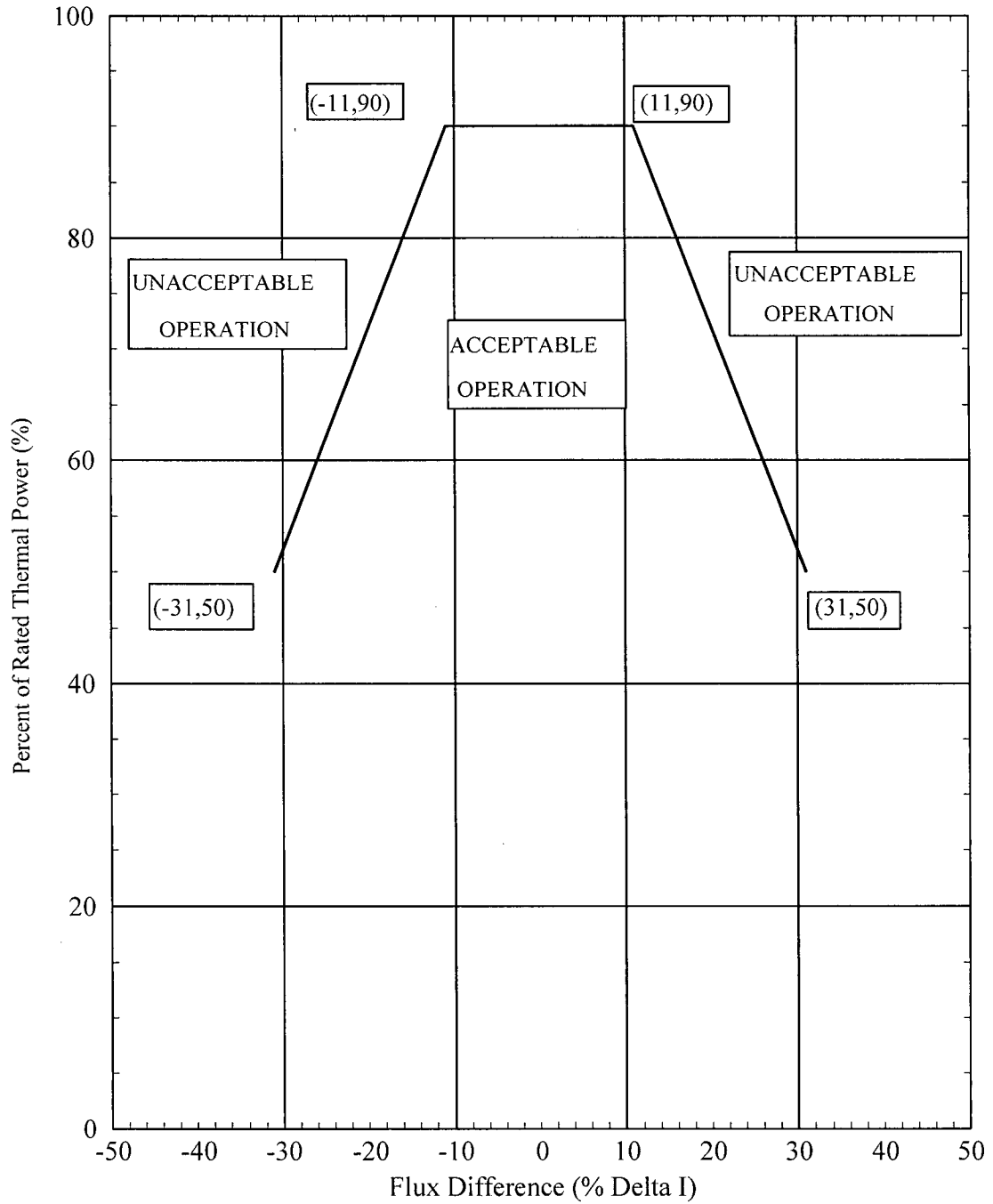


FIGURE 3

K(Z) – NORMALIZED FQ(Z) AS A FUNCTION OF CORE HEIGHT

