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NOV 05 2018

Technical Specification 6.9.1.9

LR-N18-0120

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington DC 20555-0001

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

Subject: Salem Unit 1 Core Operating Limits Report – Cycle 26, Mid-Cycle

In accordance with section 6.9.1.9 of the Salem Unit 1 Technical Specifications, PSEG Nuclear LLC submits the enclosed Core Operating Limits Report (COLR) for Salem Unit 1, Cycle 26 Mid-Cycle.

There are no commitments contained in this letter.

Should you have any questions regarding this submittal, please contact Mr. Harry Balian at (856) 339 – 2173.

Sincerely,

Patrick A. Martino
Plant Manager
Salem Generating Station

Enclosure

cc: USNRC Regional Administrator – Region 1
USNRC NRR Project Manager – Salem
USNRC Senior Resident Inspector – Salem
NJ Department of Environmental Protection, Bureau of Nuclear Engineering
Commitment Coordinator, Salem Generating Station
Corporate Commitment Coordinator, PSEG Nuclear, LLC

(The bcc list should not be submitted as part of the DCD submittal – remove this page prior to submittal and make the bcc distribution accordingly)

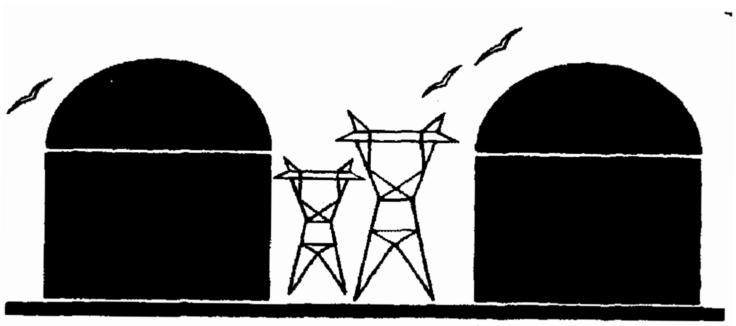
bcc: President & Chief Nuclear Officer
Site Vice President – Salem
Plant Manager – Salem
Senior Director, Regulatory Operations & Nuclear Oversight
Manager – Nuclear Oversight
Director – Regulatory Affairs
Manager – Licensing
Records Management

Enclosure

Salem Unit 1
Core Operating Limits Report (COLR) Mid-Cycle
Cycle 26

COLR SALEM 1
Revision 9
September 2018

Core Operating Limits Report for Salem Unit 1, Cycle 26



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1.0 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for Salem Unit 1 Cycle 26 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 along with the NRC-approved methodologies used to develop and/or determine COLR parameters identified in Technical Specifications.

TS Section	Technical Specifications	COLR Parameter	COLR Section	NRC Approved Methodology (Section 3.0 Number)
3.1.1.4	Moderator Temperature Coefficient	MTC	2.1	3.1, 3.6
3.1.3.5	Control Rod Insertion Limits	Control Rod Insertion Limits	2.2	3.1, 3.6
3.2.1	Axial Flux Difference	AFD	2.3	3.1, 3.2, 3.6
3.2.2	Heat Flux Hot Channel Factor - $F_Q(Z)$	$F_Q(Z)$	2.4	3.1, 3.3, 3.4, 3.5, 3.6, 3.7, 3.8
3.2.3	Nuclear Enthalpy Rise Hot Channel Factor - $F^N_{\Delta H}$	$F^N_{\Delta H}$	2.5	3.1, 3.5, 3.6, 3.7, 3.8
3.9.1	Boron Concentration	Boron Concentration	2.6	3.1, 3.6

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 $\Delta k/k/°F$.

The EOL/ARO/RTP-MTC shall be less negative than or equal to -4.4×10^{-4} $\Delta k/k/°F$.

2.1.2 The MTC Surveillance limit is:

The 300 ppm/ARO/RTP-MTC should be less negative than or equal to -3.7×10^{-4} $\Delta k/k/°F$.

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: $P = \dfrac{THERMAL\ POWER}{RATED\ THERMAL\ POWER}$

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: from BOL to 10000 MWD/MTU

F_{xy}^{RTP} =
2.03 for unrodded upper core planes 1 through 6
1.89 for unrodded upper core planes 7 through 8
1.75 for unrodded upper core planes 9 through 11
1.72 for unrodded upper core planes 12 through 13
1.73 for unrodded upper core planes 14 through 18
1.76 for unrodded upper core planes 19 through 31
1.76 for unrodded lower core planes 32 through 43
1.81 for unrodded lower core planes 44 through 48
1.88 for unrodded lower core planes 49 through 50
1.83 for unrodded lower core planes 51 through 53

1.94 for unrodded lower core planes 54 through 55
2.03 for unrodded lower core planes 56 through 61
2.07 for the core planes containing Bank D control rods

PF_{xy} = *0.3*

where: from 10000 MWD/MTU to 14000 MWD/MTU

F_{xy}^{RTP} =
2.03 for unrodded upper core planes 1 through 6
1.80 for unrodded upper core planes 7 through 8
1.73 for unrodded upper core planes 9 through 11
1.72 for unrodded upper core planes 12 through 13
1.73 for unrodded upper core planes 14 through 18
1.91 for unrodded upper core planes 19 through 31
1.96 for unrodded lower core planes 32 through 43
1.85 for unrodded lower core planes 44 through 48
1.85 for unrodded lower core planes 49 through 50
1.78 for unrodded lower core planes 51 through 53
1.82 for unrodded lower core planes 54 through 55
1.96 for unrodded lower core planes 56 through 61
2.07 for the core planes containing Bank D control rods

PF_{xy} = *0.3*

where: from 14000 MWD/MTU to EOL

F_{xy}^{RTP} =
2.00 for unrodded upper core planes 1 through 6
1.83 for unrodded upper core planes 7 through 8
1.76 for unrodded upper core planes 9 through 11
1.78 for unrodded upper core planes 12 through 13
1.80 for unrodded upper core planes 14 through 18
2.02 for unrodded upper core planes 19 through 31
2.02 for unrodded lower core planes 32 through 43
1.87 for unrodded lower core planes 44 through 48
1.85 for unrodded lower core planes 49 through 50
1.77 for unrodded lower core planes 51 through 53
1.79 for unrodded lower core planes 54 through 55
1.87 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:

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

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:

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

where:

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

U_e = Engineering uncertainty factor.
= 1.03

2.5 <u>Nuclear Enthalpy Rise Hot Channel Factor</u> - $F^N_{\Delta H}$ (Specification 3.2.3)

$$F^N_{\Delta H} = F^{RTP}_{\Delta H}\,[1.0 + PF_{\Delta H}\,(1.0 - P)]$$

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

2.5.1 $F^{RTP}_{\Delta H} = 1.65$

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, $F^N_{\Delta H}$, 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 $F^N_{\Delta H}$ shall be calculated by the following formula:

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

where: $U_{F\Delta Hm}$ = Base $F_{\Delta H}$ measurement uncertainty.
= 1.04

2.6 Boron Concentration (Specification 3.9.1)

A Mode 6 boron concentration, maintained at or above 2133 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% Δk/k uncertainty added.

b) A K_{eff} of 0.99 or less at All Rods Out (ARO), CZP conditions with a 1% Δk/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 (Westinghouse 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 (Westinghouse 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 (Westinghouse 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 (Westinghouse 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 (Westinghouse 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.

3.7 WCAP-12472-P-A, Addendum 1-A, BEACON Core Monitoring and Operations Support System, January 2000 (Westinghouse proprietary). Approved by Safety Evaluation dated September 30, 1999.

3.8 WCAP-12472-P-A, Addendum 4, BEACON Core Monitoring and Operation Support System, Addendum 4, September 2012 (Westinghouse proprietary). Approved by Safety Evaluation dated August 9, 2012.

4.0 REFERENCES

1. Salem Nuclear Generating Station Unit No. 1, up to Amendment No. 324, Renewed License No. DPR-70, Docket No. 50-272.

FIGURE 1

ROD BANK INSERTION LIMITS VS. THERMAL POWER

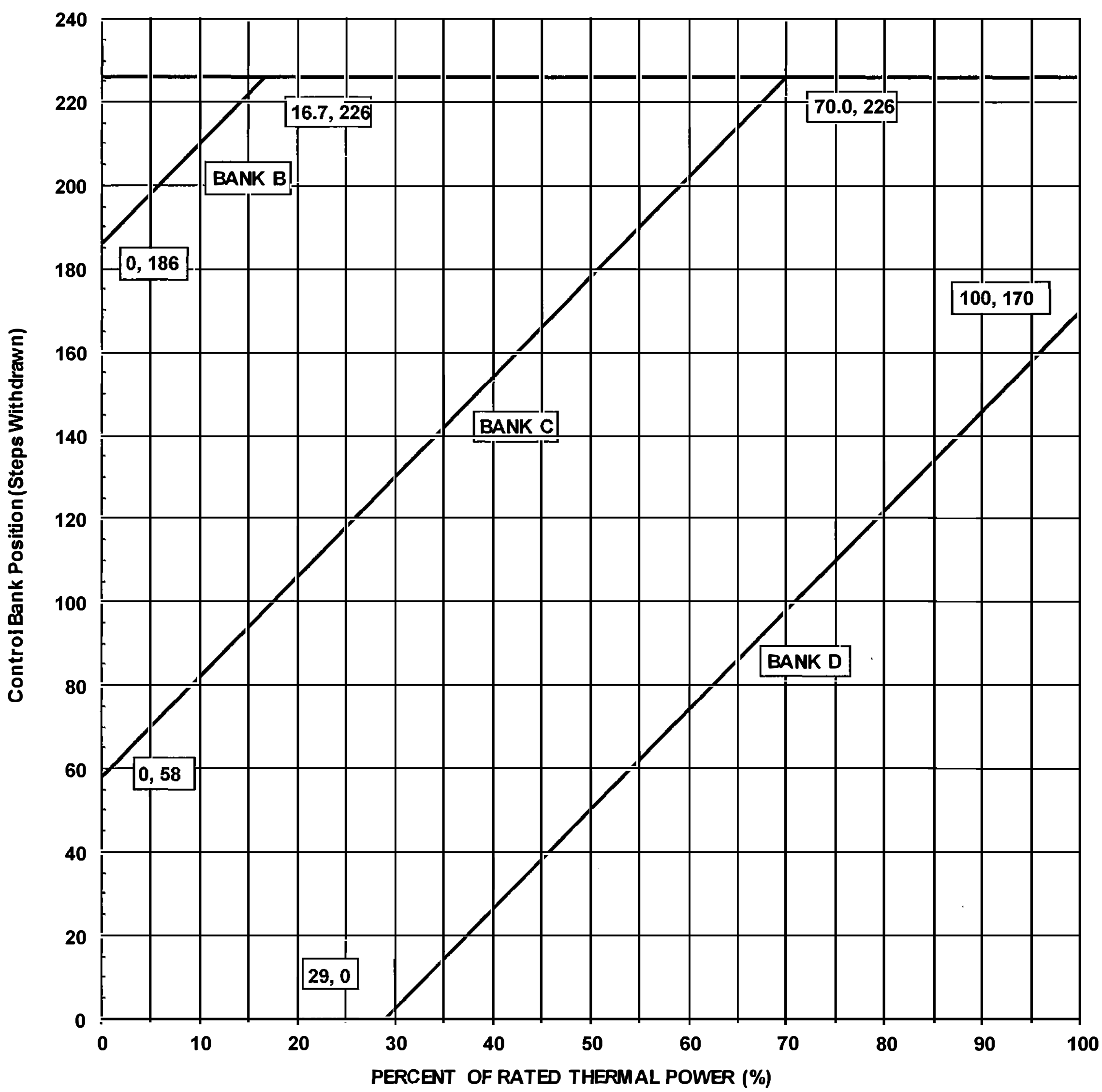

FIGURE 2

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

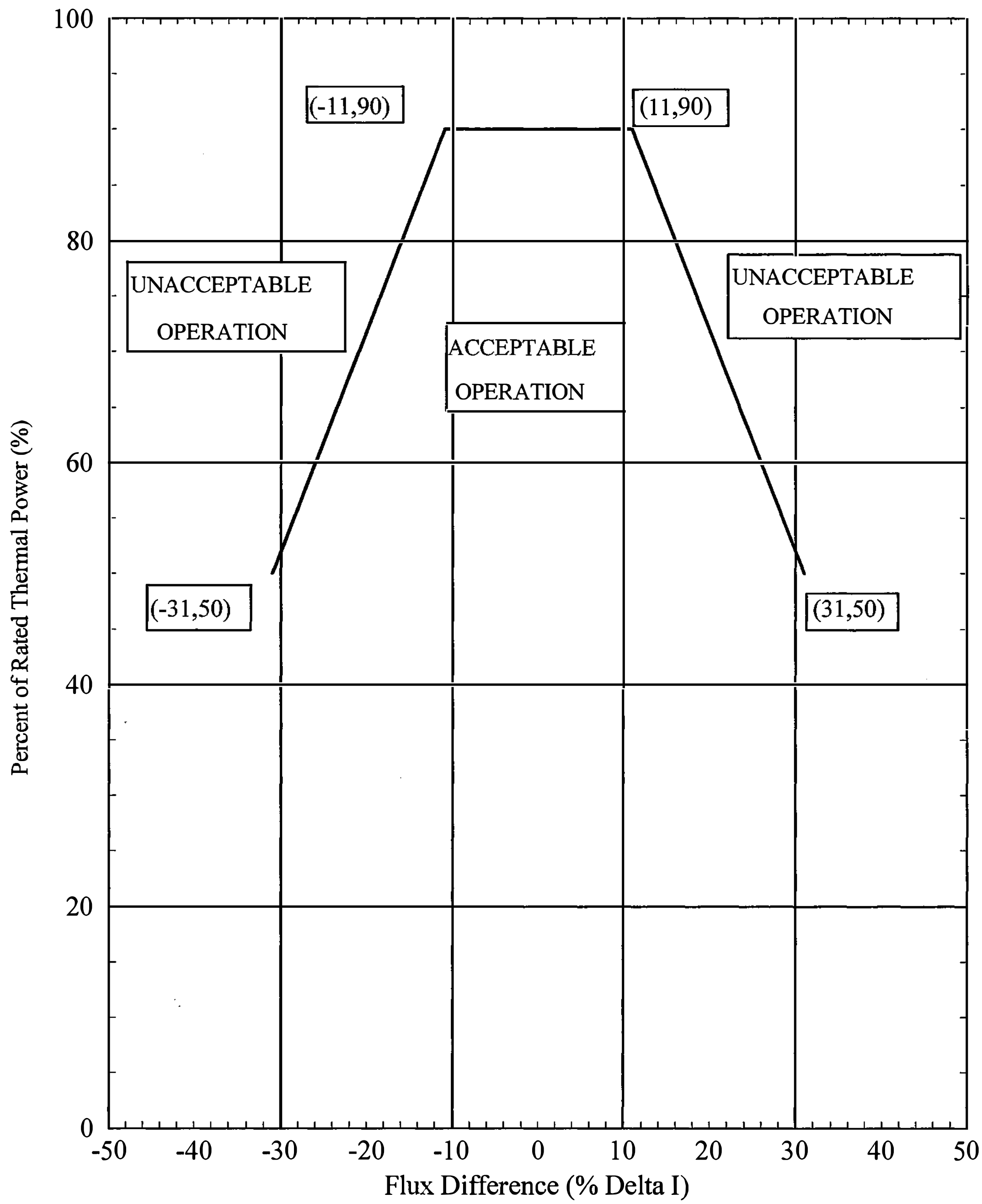

FIGURE 3

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