Mr. Harold W. Keiser
Chief Nuclear Officer & President
Nuclear Business Unit
Public Service Electric & Gas
Company
Post Office Box 236
Hancocks Bridge, NJ 08038

January 8, 1999

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NO. 2 (TAC NO. M95384)

Dear Mr. Keiser:

The Commission has issued the enclosed Amendment No. 197 to Facility Operating License No. DPR-75 for the Salem Nuclear Generating Station, Unit No. 2. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated May 10, 1996, as supplemented March 19, and August 29, 1997.

This amendment incorporates into the TSs the Margin Recovery portion of your Fuel Upgrade Margin Recovery Program and supports increased steam generator plugging, improved fuel reliability, reduced fuel costs, longer fuel cycles, reduced spent fuel storage, and enhanced reactor safety. In a letter dated November 26, 1997, the Commission issued the amendment for Salem Unit 1. In addition, several minor and nonsubstantitive clerical errors were corrected in TS 3.1.1.3 and TS 3.1.3.5 to make them consistent with the similar TSs issued for Salem Unit 1. The clerical errors consisted of several words and a punctuation mark that the licensee inadvertently did not delete when modifying the TSs for the inclusion of new insert wording.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/s/

Patrick D. Milano, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-311

Enclosures: 1. Amendment No. 197 to License No. DPR-75
2. Safety Evaluation

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OFFICE	PDI-2/PM	PDI-2/LA	EMEB	SRXB	PERB
NAME	PMilano:mw	TClark	RWessman	TCollins	CMiller
DATE	12/16/98	12/16/98	12/21/98	12/21/98	12/23/98
OFFICE	SCSB	HICB	PDI-2/D	OGC	
NAME	CBerlinger	JWermiel	~~RCapra~~	MYoung	
DATE	12/22/98	12/23/98	1/8/98	1/8/99	

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 8, 1999

Mr. Harold W. Keiser
Chief Nuclear Officer & President
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Public Service Electric & Gas
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Mr. Harold W. Keiser
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Salem Nuclear Generating Station,
Units 1 and 2

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-311

SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 197
License No. DPR-75

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for amendment filed by the Public Service Electric & Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) dated May 10, 1996, as supplemented March 19 and August 29, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-75 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 197, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

William M. Dean

William M. Dean, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: January 8, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 197

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Revise Appendix A as follows:

Remove Pages	Insert Pages
I	I
II	II
IV	IV
XI	XI
1-2	1-2
2-1	2-1
2-2	2-2
2-3	2-3
2-5	2-5
2-7	2-7
2-8	2-8
2-9	2-9
B 2-1	B 2-1
B 2-5	B 2-5
B 2-6	B 2-6
3/4 1-1	3/4 1-1
3/4 1-2	3/4 1-2
3/4 1-4	3/4 1-4
3/4 1-5	3/4 1-5
3/4 1-13	3/4 1-13
3/4 1-20	3/4 1-20
3/4 1-21	3/4 1-21
3/4 1-22	3/4 1-22
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-4	3/4 2-4
3/4 2-5	3/4 2-5
3/4 2-6	3/4 2-6
3/4 2-7	3/4 2-7
3/4 2-8	3/4 2-8
3/4 2-9	3/4 2-9
3/4 2-17	3/4 2-17
B 3/4 1-1	B 3/4 1-1
B 3/4 1-2	B 3/4 1-2
B 3/4 1-3	B 3/4 1-3
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 2-3	B 3/4 2-3
B 3/4 2-4	B 3/4 2-4

Remove Pages	Insert Pages
B 3/4 2-5	B 3/4 2-5
B 3/4 4-1	B 3/4 4-1
5-4	5-4
6-24	6-24
6-24a	6-24a

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DEFINITIONS

SECTION PAGE

1.0 DEFINITIONS

INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

1.7.1 All penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are opened under administrative control as permitted by Specification 3.6.3.

1.7.2 All equipment hatches are closed and sealed,

1.7.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3,

1.7.4 The containment leakage rates are within the limits of Specification 3.6.1.2, and

1.7.5 The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

1.8 NOT USED

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

1.9a The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.9. Unit operation within these operating limits is addressed in individual specifications.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figures 2.1-1 for 4 loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

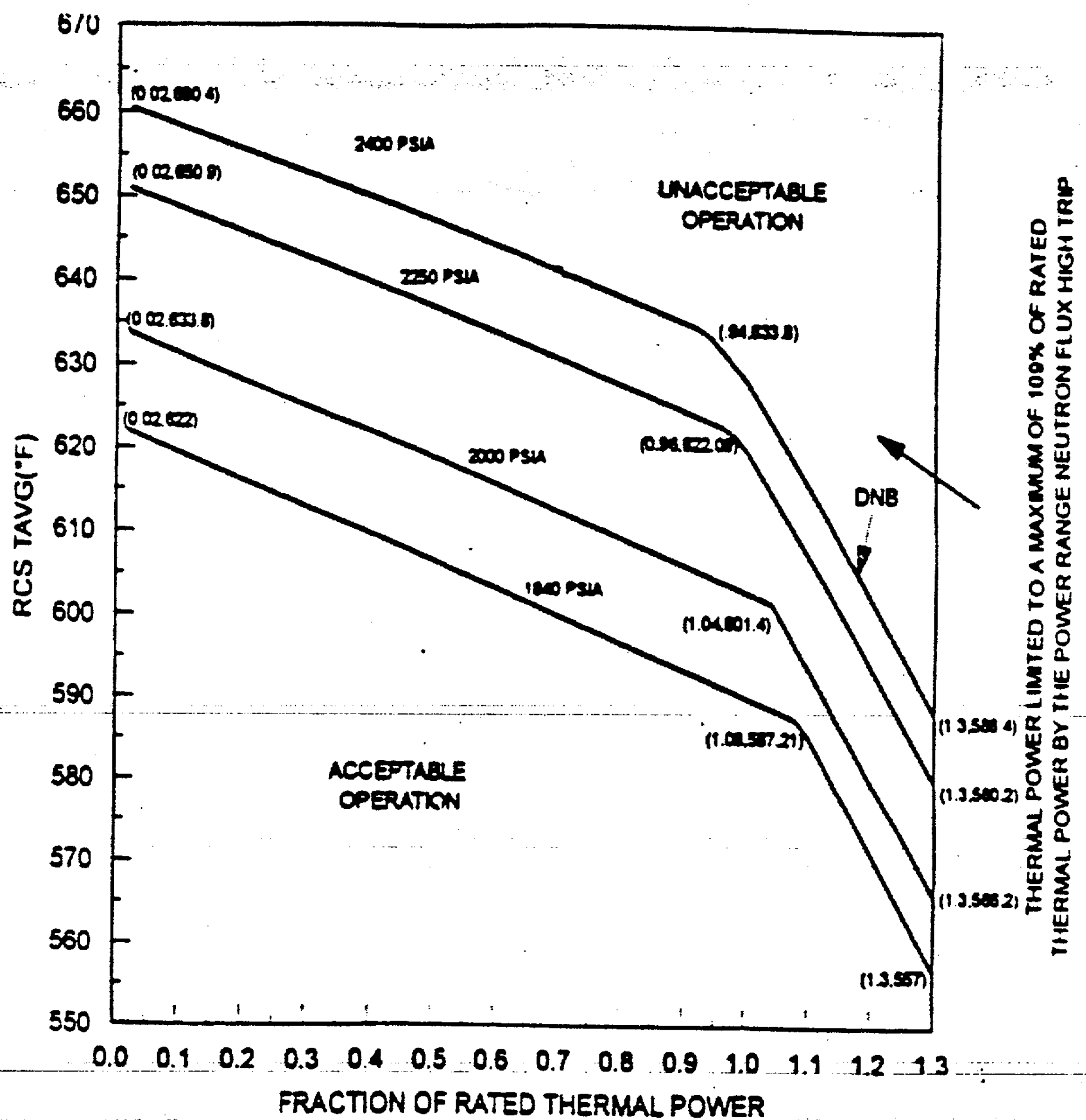

FIGURE 2.1-1 REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not applicable	Not applicable
2. Power Range, Neutron Flux	Low setpoint - ≤ 25 % of RATED THERMAL POWER High Setpoint - ≤ 109% of RATED THERMAL POWER	Low Setpoint - ≤ 26% of RATED THERMAL POWER High Setpoint - ≤ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 second	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 second
4. Power Range, Neutron Flux, High Negative Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 second	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 second
5. Intermediate Range, Neutron Flux	≤ 25% of RATED THERMAL POWER	≤ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	≤ 10^5 counts per second	≤ 1.3 x 10^5 counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 4
9. Pressurizer Pressure--Low	≥ 1865 psig	≥ 1855 psig
10. Pressurizer Pressure--High	≤ 2385 psig	≤ 2395 psig
11. Pressurizer Water Level--High	≤ 92% of instrument span	≤ 93% instrument span
12. Loss of Flow	≥ 90% of design flow per loop*	≥ 89% of design flow per loop*

*Design flow is 82,500 gpm per loop.

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: Overtemperature $\Delta T \leq \Delta T_o \left[K_1 - K_2 \frac{1+\tau_1 S}{1+\tau_2 S}(T-T') + K_3(P-P') - f_1(\Delta I)\right]$

where:

ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

T' = Indicated T_{avg} at RATED THERMAL POWER ≤ 577.9°F

P = Pressurizer pressure, psig

P' = 2235 psig (indicated RCS nominal operating pressure)

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation

τ_1 & τ_2 = Time constants utilized in the lead-lag controller for T_{avg} τ_1 = 30 secs, τ_2 = 4 secs.

S = Laplace transform operator, Sec^{-1}

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 4 Loops

K_1 = 1.22
K_2 = 0.02037
K_3 = 0.001020

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between -23 percent and +13 percent, f_1 (ΔI) = 0 (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of ($q_t - q_b$) exceeds -23 percent, the ΔT trip setpoint shall be automatically reduced by 1.26 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of ($q_t - q_b$) exceeds +13 percent, the ΔT trip setpoint shall be automatically reduced by 2.63 percent of its value at RATED THERMAL POWER.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o [K_4 - K_5 [\frac{\tau_3 S}{1+\tau_3 S}] T - K_6 (T-T") - f_2(\Delta I)]$

where: ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

$T"$ = Indicated T_{avg} at RATED THERMAL POWER $\leq$ 577.9°F

K_4 = 1.09

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature

K_6 = 0.00149/°F for $T > T"$; $K_6 = 0$ for $T \leq T"$

$\frac{\tau_3 S}{1+\tau_3 S}$ = The function generated by the rate lag controller for T_{avg} dynamic compensation

τ_3 = Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 secs.

S = Laplace transform operator, Sec^{-1}

$f_2(\Delta I)$ = 0 for all ΔI

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 1.1 percent.

Note 4: The channel's maximum trip point shall not exceed its computed trip point by more than 2.1 percent.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through correlations which have been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, decided as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is as follows: uncertainties in the WRB-1 and WRB-2 correlations, plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and computer codes are considered statistically such that there is at least a, 95 percent probability with 95 percent confidence level that DNBR will not occur on the most limiting fuel rod during Condition I and II events. This establishes a design DNBR value which must be met in plant safety analyses using values of input Parameters without uncertainties.

The curves of Figure 2.1-1 shows the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the design DNBR value, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

The curves are based on an enthalpy hot channel factor, $F^{RTP}_{\Delta H}$ and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^{N}_{\Delta H}$ at reduced power based on the expression:

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

Where: $F^{RTP}_{\Delta H}$ is the limit at RATED THERMAL POWER in the Core Operating Limits Report (COLR).

$PF_{\Delta H}$ is the Power Factor Multiplier for $F^{N}_{\Delta H}$ specified in the COLR, and P is $\dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

These limiting heat flux conditions are higher than those calculated for the range of all control rod positions from FULLY WITHDRAWN to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (delta I) function of the Overtemperature trip. When the axial power

Operation with a reactor coolant loop out of service below the 4 loop P-8 setpoint does not require reactor protection system setpoint modification because the P-8 setpoint and associated trip will prevent DNB during 3 loop operation exclusive of the Overtemperature delta T setpoint. Three loop operation above the 4 loop P-8 has not been evaluated and is not permitted.

Overpower Delta T

The Overpower delta T reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

LIMITING SAFETY SYSTEM SETTINGS

BASES

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 11 percent of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drop below 90% of nominal full loop flow. Above 36% (P-8) of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. This latter trip will prevent the minimum value of the DNBR from going below the design DNBR value during normal operational transients.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays of the auxiliary feedwater system.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} > 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.3% delta k/k.

APPLICABILITY: MODES 1, 2*, 3, and 4.

ACTION:

With the SHUTDOWN MARGIN less than 1.3% delta k/k, immediately initiate and continue boration at ≥33 gpm of a solution containing ≥ 6,560 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.3% delta k/k:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits in the COLR per Specification 3.1.3.5.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits in the COLR per Specification 3.1.3.5.

*See Special Test Exception 3.10.1

SURVEILLANCE REQUIREMENTS (Continued)

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit in the COLR per Specification 3.1.3.5.

e. When in MODES 3 or 4, at least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. Control rod position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the limits specified in the CORE OPERATING LIMITS REPORT (COLR). The maximum upper limit shall be less positive than or equal to 0 $\Delta k/k/°F$.

APPLICABILITY: Beginning of Cycle Life (BOL) Limit - MODES 1 and 2* only#
End of Cycle Life (EOL) Limit - MODES 1, 2 and 3 only#

ACTION:

a. With the MTC more positive than the BOL limit specified in the COLR, operations in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the BOL limit specified in the COLR within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits in the COLR per Specification 3.1.3.5.

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

3. In lieu of any other report required by Specification ~~6.9.1, a~~ Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the EOL limit specified in the COLR, be in HOT SHUTDOWN within 12 hours.

*With K_{eff} greater than or equal to 1.0

#See Special Test Exception 3.10.3

MODERATOR TEMPERATURE COEFFICIENT

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit specified in the COLR prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. The MTC shall be measured at any THERMAL POWER and compared to the 300 ppm surveillance limit specified in the COLR (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than the 300 ppm surveillance limit specified in the COLR, the MTC shall be remeasured, and compared to the EOL MTC limit specified in the COLR at least once per 14 EFPD during the remainder of the fuel cycle.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods, shall be OPERABLE and positioned within ± 18 steps (indicated position) when reactor power is ≤ 85% RATED THERMAL POWER, or ± 12 steps (indicated position) when reactor power is > 85% RATED THERMAL POWER, of their group step counter demand position within one hour after rod motion.

APPLICABILITY: MODES 1* and 2*

ACTION:

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one full length rod inoperable or mis-aligned from the group step counter demand position by more than ± 18 steps (indicated position) at ≤ 85% RATED THERMAL POWER or ± 12 steps (indicated position) at > 85% RATED THERMAL POWER, be in HOT STANDBY within 6 hours.

c. With one full length rod inoperable due to causes other than addressed by ACTION a, above, or mis-aligned from its group step counter demand position by more than ± 18 steps (indicated position) at ≤ 85% RATED THERMAL POWER or ± 12 steps (indicated position) at > 85% RATED THERMAL POWER, POWER OPERATION may continue provided that within one hour either:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The remainder of the rods in the bank with the inoperable rod are aligned to within ± 18 steps (indicated position) at ≤ 85% RATED THERMAL POWER or ± 12 steps (indicated position) at >85% RATED THERMAL POWER, of the inoperable rod while maintaining the rod sequence and insertion limits in the COLR per Specification 3.1.3.5. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.5 during subsequent operation, or

   3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

*See Special Test Exceptions 3.10.2 and 3.10.3.

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEM SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.5 The control banks shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1*, and 2*#

ACTION:

With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

a. Restore the control banks to within the limits within two hours, or

b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the insertion limits specified in the COLR, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.5 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours by use of the group demand counters and verified by the analog rod position indicators** except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours**.

* See Special Test Exceptions 3.10.2 and 3.10.3

**For power levels below 50% one hour thermal "soak time" is permitted. During this soak time, the absolute value of rod motion is limited to six steps.

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3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE shall be maintained within the target band about the target flux difference as specified in the CORE OPERATING LIMITS REPORT (CORL).

APPLICABILITY: MODE 1 ABOVE 50% RATED THERMAL POWER*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the target band about the target flux difference as specified in the COLR 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 target band as specified in the COLR for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

2) The indicated AFD is within the limits as specified in the COLR. 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 less than or equal 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 as specified in the COLR. 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 target band as specified in the COLR 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 target band as specified in the COLR for more than 1 hour penalty deviation cumulative during the previous 24 hours. Power increases above 50% of RATED THERMAL POWER do not require being within the target band provided the accumulative penalty deviation is not violated.

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 target band when at least 2 or more OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each one minute of POWER OPERATION outside of the 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 target band at THERMAL POWER levels below 50% of RATED THERMAL POWER.

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3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(z) \leq \dfrac{F_Q^{RTP}}{P} * K(z)$ for $P > 0.5$, and

$F_Q(z) \leq \dfrac{F_Q^{RTP}}{0.5} * K(z)$ for $P > 0.5$, and

Where F_Q^{RTP} = the F_Q limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR),

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

$K(z)$ = the normalized $F_Q(z)$ as a function of core height as specified in the COLR.

APPLICABILITY: MODE 1

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower delta T Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower delta T Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a. above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Comparing the F_{xy} computed (F^C_{xy}) obtained in b, above to:

1. The F_{xy} limits for RATED THERMAL POWER (F^{RTP}_{xy}) for the appropriate measured core planes given in e. and f., below, and

2. The relationship:

$$F^L_{xy} = F^{RTP}_{xy} \; [1 + PF_{xy}(1-P)]$$

where F^L_{xy} is the limit for fractional THERMAL POWER operation expressed as a function of F^{RTP}_{xy}, PF_{xy} is the power factor multiplier for F_{xy} in the CORL, and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

1. When F^C_{xy} is greater than the F^{RTP}_{xy} limit for the appropriate measured core plane but less than the F^L_{xy} relationship, additional power distribution maps shall be taken and F^C_{xy} compared to F^{RTP}_{xy} and F^L_{xy} :

a) Either within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F^C_{xy} was last determined, or

SURVEILLANCE REQUIREMENTS (Continued)

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

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

e. The F_{xy} limit for Rated Thermal Power (F^{RTP}_{xy}) shall be provided for all core planes containing bank "D" control rods and all unrodded core planes in the COLR per specification 6.9.1.9.

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% ± 2%, 32.1% ± 2%, 46.4% ± 2%, 60.6% ± 2% and 74.9% ± 2%, inclusive.
4. Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the bank "D" 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^{C}_{xy} exceeds F^{L}_{xy}.

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.

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POWER DISTRIBUTION LIMITS

3/4.2.3 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:

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

Where $F^{RTP}_{\Delta H}$ is the limit at RATED THERMAL POWER in the Core Operating Limits Report (COLR).

$PF_{\Delta H}$ is the Power Factor Multiplier for $F^N_{\Delta H}$ specified in the COLR, and P is $\dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

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.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 4 Loops in Operation
Reactor Coolant System T_{avg}	≤ 582.9°F
Pressurizer Pressure	≥ 2200 psia*
Reactor Coolant System Total Flow Rate	≥ 341,000 gpm#

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* Limit not applicable during either a THERMAL POWER ramp in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% RATED THERMAL POWER.

Includes a 2.4% flow uncertainty plus a 0.1% measurement uncertainty due to feedwater venturi fouling.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.3% $\Delta k/k$ is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than or equal to 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% $\Delta k/k$ shutdown margin provides adequate protection.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the accident and transient analyses.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (MTC) (Continued)

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analysis to nominal operating conditions. These corrections involved: (1) a conversion of the MDC used in the FSAR analysis to its equivalent MTC, based on the rate of change of moderator density with temperature at RATED THERMAL POWER conditions, and (2) subtracting from this value the largest differences in MTC observed between EOL, all rods withdrawn, RATED THERMAL POWER conditions, and those most adverse conditions of moderator temperature and pressure, rod insertion, axial power skewing, and xenon concentration that can occur in normal operation and lead to a significantly more negative EOL MTC at RATED THERMAL POWER. These corrections transformed the MDC value used in the FSAR analysis into the limiting End Of Cycle Life (EOL) MTC value. The 300 ppm surveillance limit MTC value represents a conservative value at a core condition of 300 ppm equilibrium boron concentration that is obtained by correcting the limiting EOL MTC for burnup and born concentration.

The surveillance requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains with its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the P-12 interlock is above its allowable setpoint, 4) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 5) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid transfer pumps, and 5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature ≥ 350°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.3% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability (minimum boration volume) requirement is established to conservatively bound expected operating conditions throughout core operating life. The analysis assumes that the most reactive control rod is not inserted into the core. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires borated water from a boric acid tank in accordance with TS Figure 3.1-2, and additional makeup from either: (1) the second boric acid tank and/or batching, or (2) a maximum of 41,800 gallons of 2,300 ppm borated water from the refueling water storage tank. With the refueling water storage tank as the only borated water source, a maximum of 73,800 gallons of 2,300 ppm borated water is required. However, to be consistent with the ECCS requirements, the RWST is required to have a minimum contained volume of 350,000 gallons during operations in MODES 1, 2, 3 and 4.

The boric acid tanks, pumps, valves, and piping contain a boric acid solution concentration of between 3.75% and 4% by weight. To ensure that the boric acid remains in solution, the tank fluid temperature and the process pipe wall temperatures are monitored to ensure a temperature of 63°F, or above is maintained. The tank fluid and pipe wall temperatures are monitored in the main control room. A 5°F margin is provided to ensure the boron will not precipitate out.

Should ambient temperature decrease below 63°F, the boric acid tank heaters, in conjunction with boric acid pump recirculation, are capable of maintaining the boric acid in the tank and in the pump at or about 63°F. A small amount of boric acid in the flowpath between the boric acid recirculation line and the suction line to the charging pump will precipitate out, but it will not cause flow blockage even with temperatures below 50°F.

With the RCS temperature below 350°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE OPERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

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) meeting the DNB Design Criteria 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.

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

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope of the F_Q limit specified in the CORE OPERATING LIMITS REPORT (COLR) 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.

POWER DISTRIBUTION LIMITS

BASES

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the target band in the COLR per Specification 3.2.1 about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits specified in the COLR while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of rated THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD are derived from the plant nuclear instrumentation system through the AFD Monitor Alarm. A control room recorder continuously displays the auctioneered high flux difference and the target band limits as a function of power level. An alarm is received any time the auctioneered high flux difference exceeds the target band limits. Time outside the target band is graphically presented on the strip chart.

Figure B 3/4 2-1 shows a typical monthly target band.

INFORMATION ONLY*

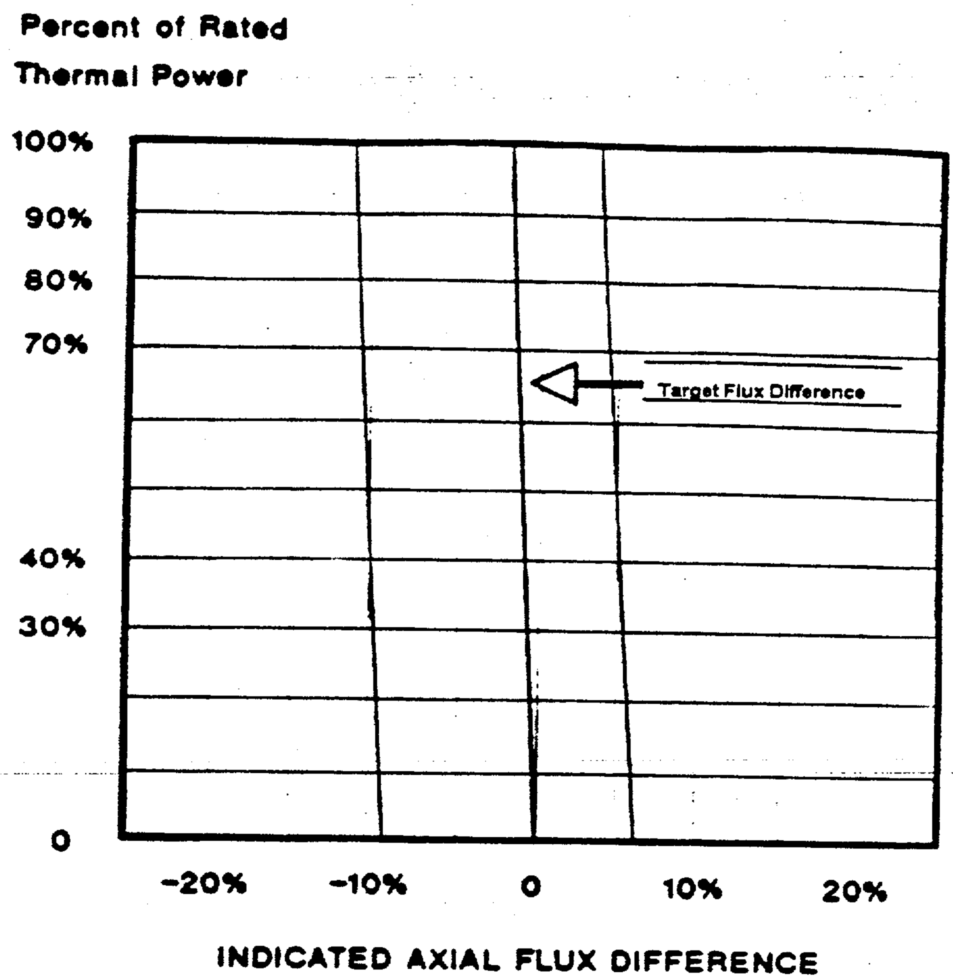

Figure B 3/4 2-1 TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

* REFER TO COLR FIGURE 2 FOR ACTUAL LIMITS

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

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 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 limits are maintained provided:

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

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.

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 through d above, are maintained.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent 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 $F^N_{\Delta H} \leq F^{RPT}_{\Delta H}/1.08$. Where $F^{RPT}_{\Delta H}$ is the limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR). The 8% allowance is based on the following considerations:

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

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 test can be compensated for in F_Q by restricting axial flux distributions. This compensation for $F^N_{\Delta H}$ is less rapidly available.

The radial peaking factor $F_{xy}(Z)$ is measured periodically to provide assurance that the hot channel factor $F_Q(Z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER F^{RTP}_{xy} , as provided in COLR per specification 6.9.1.9, 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.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation, meet the DNB design criteria during all normal operations and anticipated transients. In MODES 1 and 2 with less than all coolant loops in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal for removing decay heat; but, single failure considerations require all loops be in operation whenever the rod control system is energized and at least one loop be in operation when the rod control system is deenergized.

In MODE 4, a single reactor coolant loop or RHR loop provides sufficient heat removal for removing decay heat; but, single failure considerations require that at least 2 loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires that two RHR loops be OPERABLE.

In MODE 5, single failure considerations require that two RHR loops be OPERABLE. For support systems: Service Water (SW) and Component Cooling (CC), component redundancy is necessary to ensure no single active component failure will cause the loss of Decay Heat Removal. One piping path of SW and CC is adequate when it supports both RHR loops. The support systems needed before entering into the desired configuration (e.g., one service water loop out for maintenance in Modes 5 and 6) are controlled by procedures, and include the following:

- A requirement that two RHR, two CC and two SW pumps, powered from two different vital buses be kept operable
- A listing of the active (air/motor operated) valves in the affected flow path to be locked open or disabled

Note that four filled reactor coolant loops, with at least two steam generators with at least their secondary side water level greater than or equal to 5% (narrow range), may be substituted for one residual heat removal loop. This ensures that a single failure does not cause a loss of decay heat removal.

The operation of one Reactor Coolant Pump or one RHR Pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during Boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with Boron concentration reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump below P-7 with one or more RCS cold legs less than or equal to 312°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer (thereby providing a volume into which the primary coolant can expand, or (2) by restricting the starting of Reactor Coolant Pumps to those times when secondary water temperature in each steam generator is less than 50°F above each of the RCS cold leg temperatures.

DESIGN FEATURES

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment is designed and shall be maintained for a maximum internal pressure of 47 psig. Containment air temperatures up to 351.3°F are acceptable providing the containment pressure is in accordance with that described in the UFSAR.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of zircaloy or ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. 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.

5.4 REACTOR COOLANT SYSTEM

DESIGN FEATURE AND TEMPERATURE

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

a. In accordance with the code requirement specified in Section 4.1 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 12,446 ± 426 cubic feet at a nominal T_{avg} of 573.0°F.

d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent or absorbent (e.g., cement, urea formaldehyde).

The Radioactive Effluent Release Reports shall include a list of descriptions of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

6.9.1.9 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Moderator Temperature Coefficient Beginning of Life (BOL) and End of Life (EOL) limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

2. Control Bank Insertion Limits for Specification 3/4.1.3.5,

3. Axial Flux Difference Limits and target band for Specification 3/4.2.1,

4. Heat Flux Hot Channel Factor, F_Q, its variation with core height, K(z), and Power Factor Multiplier PF_{xy}, Specification 3/4.2.2, and

5. Nuclear Enthalpy Hot Channel Factor, and Power Factor Multiplier, $PF_{\Delta H}$ for Specification 3/4.2.3.

b. 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:

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

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. WCAP-10054-P-A, Rev. 1, Westinghouse Small Break ECCS Evaluation Model Using 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.

4. WCAP-10266-P-A, Rev. 2, The 1981 Version of Westinghouse Evaluation Model Using BASH Code, Rev. 2. March 1987 (W Proprietary) Methodology for Specification 3/4.2.2 Heat Flux Hot Channel Factor. Approved by Safety Evaluation dated November 13, 1986.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid-cycle revisions or supplements shall be provided upon issuance for each reload cycle to the NRC.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the Administrator, USNRC Region I within the time period specified for each report.

6.9.3 Violations of the requirements of the fire protection program described in the Updated Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire shall be submitted to the U. S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator of the Regional Office of the NRC via the Licensee Event Report System within 30 days.

6.9.4 When a report is required by ACTION 8 OR 9 of Table 3.3-11 "Accident Monitoring Instrumentation", a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring for inadequate core cooling, the cause of the inoperability, and the plans and schedule for restoring the instrument channels to OPERABLE status.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 197 TO FACILITY OPERATING LICENSE NO. DPR-75

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

DOCKET NO. 50-311

1.0 INTRODUCTION

By letter dated May 10, 1996, as supplemented March 19 and August 29, 1997, the Public Service Electric & Gas Company (the licensee) submitted a request for changes to the Salem Nuclear Generating Station, Unit Nos. 1 and 2, Technical Specifications (TSs). The requested changes would incorporate into the TSs the Margin Recovery portion of the licensee's Fuel Upgrade Margin Recovery Program and support increased steam generator plugging, improved fuel reliability, reduced fuel costs, longer fuel cycles, reduced spent fuel pool storage, and enhanced reactor safety. The Fuel Upgrade portion, which involved the use of VANTAGE+ fuel and ZIRLO cladding, was approved in Amendments 154/135, dated August 22, 1994. The March 19 and August 29, 1997, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

On November 27, 1997, the NRC issued Amendment No. 201 for Salem Unit 1 which incorporated the requested changes. The licensee had requested that this amendment not be implemented on Salem Unit 2 until the next refueling outage, which is scheduled to begin in April 1999. In order to reduce the likelihood of an administrative error, the staff has decided not to issue the amendment for Salem Unit 2 at that time but instead issue it closer to when it will be implemented.

2.0 EVALUATION

In its May 10, 1996, letter, the licensee requested changes to the TSs to support the Margin Recovery Program. The proposed changes to the Salem TSs would (1) relocate cycle-specific parameter limits from the TSs to the Core Operating Limits Report (COLR), (2) eliminate those requirements associated with three-loop operation, (3) reduce the required reactor coolant system (RCS) flow for the low flow reactor trip setpoint, (4) revise the reactor core safety limits and the equations for calculating the Overtemperature Delta Temperature and the Overpower Delta Temperature trip setpoints, (5) revise the TS Bases for the Safety Limits, (6) change the required shutdown margin in Modes 1 (power operation) through 4 (hot shutdown), (7) revise the departure from nucleate boiling parameters of RCS $T_{average}$, pressurizer pressure, and RCS flow, and (8) make some editorial changes and clarifications. The specific changes are described in addition detail in the following sections.

2.1 Nuclear Design

The licensee analyzed the effects of the Margin Recovery Program (MRP) and the associated TS changes on the nuclear design bases and methodologies for Salem, Units 1 and 2. Plant-specific TSs impacting the nuclear design bases were also reviewed. The review resulted in identifying axial and radial peaking factors as well as shutdown margin limits that could impact the design bases.

The increased peaking factor limits and the reduced shutdown margin requirements will increase fuel management flexibility by placing additional burned fuel on the periphery of the core, leading to lower neutron leakage and increased fuel economy.

Typical cycle-to-cycle variations in core loading patterns, as well as normal methods of feed enrichment variation and insertion of burnable absorbers, will be used to control peaking factors, and for assuring compliance with peaking factors TSs.

The implementation of the MRP TS changes will not affect the nuclear design philosophy or the associated methodology. The reload design philosophy includes the evaluation of the reload core physics safety parameters. The reload design is comprised of the reanalyzed nuclear design input parameters to the Final Safety Analyses Report (FSAR) safety evaluation for each reload cycle. These key safety parameters will be reevaluated for each reload cycle at Salem, Units 1 and 2. If one or more of the input parameters falls outside the bounds typically assumed in the safety analysis, the affected transients will be reevaluated and/or reanalyzed, and the results will be documented in the revised safety evaluation (RSE) for that cycle and Unit. Therefore, the NRC staff finds the results acceptable.

2.2 Thermal and Hydraulic Design

The departure from nuclear boiling (DNB) analysis submitted by the licensee incorporates the plant-specific revised thermal design procedure (RTDP, WCAP-13651, "Westinghouse Revised Thermal Design Procedure - Instrument Uncertainty Methodology, Salem Units 1 & 2," August 1993) and an improved computer model called THINC-IV. The licensee noted that the W-3 correlation and the Standard Thermal Design Procedure (STDP) are still used when conditions are outside the range of the WRB-1 correlation and the RTDP. The MRP is a consequence of the significant improvements in the accuracy of the critical heat flux predictions over previous DNB correlations. Specific plant parameters, DNB correlation predictions, and fuel fabrication parameters are combined statistically to obtain the overall DNB uncertainty factor typically used to satisfy the DNB ratio (DNBR) 95/95-percent design criterion for any Condition I or II event.

When the licensee performed its DNB safety analyses, it increased the DNBR limit to gain a DNB margin capable of offsetting the effects of rod bow, transient core and any other DNB penalties that may occur, and to gain flexibility in design and operation of the plant. The DNBR limit values of 1.34 for the typical cells and 1.33 for the thimble cells were used in the safety analysis.

The increase in the DNB margin gained through the RTDP methodology with the WRB-1 correlation led to the request for the increase in the full power radial peaking factor $F_{\Delta H}$ from 1.55 to 1.65. All remaining thermal-hydraulic design criteria were also satisfied in the safety analyses. Therefore, the NRC staff finds the results of this analysis acceptable.

2.3 Accident Analysis

TSs 3/4.1, 3/4.2 and their associated bases affected by the MRP are those pertaining to the radial peaking factor Ft, and the total peaking factor F_Q. Analyses conducted by the licensee led to an increase in the radial peaking factor to 1.65 and an increase in the total peaking factor to 2.40. The accidents affected by these increases are the rod withdrawal from subcritical, the dropped rod, partial loss of forced reactor coolant flow, complete loss of forced reactor coolant flow, locked rotor, single rod cluster control assembly (RCCA) withdrawal at power, small-break loss of coolant accident (LOCA), and large-break LOCA. The most limiting transients are the complete loss of forced reactor coolant flow and the large break LOCA.

2.3.1 Partial and Complete Loss of Coolant Flow

The licensee reviewed the partial and complete loss-of-coolant transient accident for Salem, Units 1 and 2, using Nuclear Regulatory Commission (NRC) approved computer codes and methods. The analysis bounded operation with steam generator tube plugging levels up to (1) a uniform steam generator tube plugging level of 20 percent and (2) asymmetric steam generator tube plugging conditions with an average steam generator tube plugging level of 20 percent and a maximum steam generator tube plugging level of 25 percent in any steam generator.

Data submitted by the licensee showed that for the partial loss-of-flow event, the DNBR does not decrease below the safety analysis limit value at any time during the transient. The same analysis also showed that the DNBR is always greater than the more limiting DNBR calculated for the "complete loss-of-flow" event.

2.3.2 Large Break LOCA

The licensee analyzed the large break LOCA for Salem Units 1 and 2 applicable for the MRP utilizing a modified version of the NRC-approved 1981 Evaluation Model with BASH methodology and computer codes. Typically, these documents describe the major phenomena modeled, the interface between the computer codes, and the features of the codes that ensure compliance with the requirements defined in Appendix K to 10 CFR Part 50. The codes in question are used to assess the core heat transfer characteristics and to determine if the core remains susceptible to cooling throughout the blowdown, refill, and reflood phases of the LOCA.

The assumptions used in the analysis were plant-specific and reflected the requested changes, i.e., the changes in the peaking factors, shutdown margin, steam generator tube plugging, etc.

The basis for the analysis was the limiting double-ended guillotine break of the reactor coolant system (RCS) cold leg. The emergency core cooling system (ECCS) will conform to the acceptance criteria of 10 CFR 50.46 as follows:

a) The calculated peak fuel element cladding temperature does not exceed 2200 °F.

b) The amount of fuel element cladding that reacts chemically with water or steam does not exceed one percent of the total amount of Zircaloy in the reactor.

c) The localized cladding oxidation limit of 17 percent is not exceeded during or after quenching.

d) The core remains amenable to cooling during and after the break.

e) The core temperature is reduced and decay heat is removed for an extended period of time. This is required for removing the heat from the long-lived radioactivity in the core.

The LOCA analysis resulted in a peak cladding temperature of 2020 °F for the limiting break case. The analysis also indicated that the cladding temperature began to decrease at a time when the core geometry was still amenable to cooling. The licensee has shown in this submittal that the large break ECCS analysis (as conducted) results in compliance with the requirements of 10 CFR 50.46. The staff reviewed each of the transients affected by the TS changes noted above and finds the results acceptable.

2.3.3 Overtemperature and Overpower Delta T

The overtemperature and overpower delta trip (OT/OPDT) function K values in TS Table 2.2-1 are revised to reflect the fuel upgrade/MRP based on the most conservative core limits. The most conservative core limits were based on the RTDP safety limits. The core limits used to calculate the OT/OPDT setpoints were given in Table 4.1-1 of the submittal. The licensee reanalyzed the updated Final Safety Analysis Report (UFSAR) events that rely on the OT/OPDT for protection, to reflect the setpoint changes in the revised TS.

The licensee confirmed through analysis that the new OT/OPDT setpoints protect the core safety limits. Therefore, the proposed changes are acceptable.

2.3.4 Shutdown Margin

The minimum required shutdown margin in Modes 1 through 4 is being changed from 1.6 percent delta k/k to 1.3 percent delta k/k. This reduction is due to the implementation of the

MRP and is supported by the design-basis safety analysis provided in the current submittal. The licensee reanalyzed the pertinent transients affected by this reduction in the shutdown margin, such as the credible steam line break (CSLB) and the main steamline break (MSLB). The MSLB is the most limiting of the two transients, and is classified as a Condition IV event.

An MSLB depicts a rupture in the main steam pipe, which will result in an initial increase in steam flow, which decreases during the accident as the steam pressure falls. The licensee performed the analysis to determine such parameters as core heat flux, RCS temperature, and pressure resulting from cooldown following a steamline break. Computer codes such as LOFTRAN and THINC were used to determine these parameters as well as the DNBR. The staff reviewed the assumed conditions that existed at the time of the MSLB and found the analysis acceptable.

The analysis showed that the previous steamline break analyses would not be significantly affected by the MRP implementation and that all the cases that were reanalyzed continue to produce acceptable results. The analysis also indicated that the previously limiting case (complete severance of a pipe inside the containment) remains the limiting event and bounds the results of the other steamline break and the main steam system (MSS) depressurization cases. The DNB analysis for the limiting case was determined to be limiting with respect to minimum margin to DNB, that is, the minimum DNBR remains above the safety limit, and that the limiting case bounds the other steamline break core response results. The staff find this conclusion acceptable.

2.3.5 Moderator Temperature Coefficient

The licensee reanalyzed the accident events associated with the moderator temperature coefficient (MTC) in support of the implementation of the MRP. The staff reviewed each of the affected transients, in particular the limiting transients which are the feedwater malfunction (FWM) and the MSLB analyzed above. The licensee analyzed the feedwater malfunction cases using the most conservative assumptions.

The analysis indicated that the decrease in the feedwater temperature transient due to an opening in the low-pressure feedwater heater bypass valve is less severe (less limiting) than the excessive load increase event, described in Section 4.1.11 of the submittal. The licensee reanalyzed the excessive load increase event as described in Section 4.4.11 and, on the basis of the results presented in that section, the applicable acceptance criteria for the decrease in the feedwater temperature event have been met.

Alternatively, the feedwater flow at full-power transient results indicate that the DNBR values are above the safety analysis limit value. Further analysis conducted at hot zero power showed that the minimum DNBR remains above the safety analysis limit for a maximum reactivity insertion rate. This result conservatively bounds the excessive feedwater addition at no-load conditions. The staff finds these results acceptable.

The licensee reanalyzed all the events associated with Reactor Coolant System (RCS) flow and with increased pressure and temperature uncertainty, using approved codes and methodology pertinent to each individual event. For the most limiting case, the complete loss of flow event, the analysis showed that the DNBR does not decrease below the limit value at any time during the transient. The staff finds the results acceptable.

2.4 Core Operating Limit Report

The licensee has requested the establishment of a Core Operating Limit Report (COLR) for Salem Nuclear Generating Station, Units I and 2. In establishing the COLR, the licensee utilized the NRC guidance for establishing a COLR to control cycle-specific limits, as stated in Generic Letter 88-16, "Removal of Cycle Specific Parameter Limits for Technical Specifications," dated October 4, 1988. The COLR will be updated and submitted to the NRC with each fuel cycle, including mid-cycle revisions to the fuel cycle. Cycle-specific limits for

Salem Unit 1 Cycle 13 have been prepared in accordance with the requirements of TS 6.9.1.9. The TSs affected are listed below:

3/4.1.1.4	Moderator Temperature Coefficient
3/4.1.3.5	Control Rod Insertion Limits
3/4.2.1	Axial Flux Difference
3/4.2.2	Heat Flux Hot Channel Factor
3/4.2.3	Nuclear Enthalpy Hot Channel Factor

The core operating limits will be established before each reload cycle, or before any portion of a reload cycle, and will be documented in the COLR. The analytical methods used to determine the core operating limits will be those previously reviewed and approved by the NRC as listed below (and in proposed TS 6.9.1.9.b).

a. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 (W Proprietary).
b. WCAP-8385, "Power Distribution Control and Load Following Procedures - Topical Report," September 1974 (W Proprietary).
c. WCAP-10054-P-A, Rev. 1, "Westinghouse Small Break ECCS Evaluation Model Using NOTRUMP Code," August 1985 (W Proprietary).
d. WCAP-10266-P-A, Rev. 2, "The 1981 Version of Westinghouse Evaluation Model Using BASH Code," March 1987 (W Proprietary).

These methodologies are appropriate for use at Salem and will ensure that the core operating limits will be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal-hydraulics limits, ECCS limits, nuclear limits such as shutdown margin (SDM), transient analysis limits, and accident analysis limits) of the safety analysis are met. Therefore, removal of these cycle-specific paramenters is consistent with 10 CFR 50.36 as the listing of methodologies provides adequate controls to provide assurance of safe operation. The proposed amendment also states that the COLR, including any mid-cycle revisions or supplements thereto, shall be sent upon issuance for each reload cycle to the NRC. The NRC staff finds these TS changes acceptable.

2.5 Instrumentation Uncertainty Methodology

The DNB analysis of the core incorporates the RTDP described in WCAP-13651. The RTDP uncertainties are combined statistically to obtain the overall DNBR uncertainty factor such that the probability that DNB will not occur on the most limiting fuel rod is at least 95% (at a 95% confidence level) for any Condition I or II event. The above probability is based on the assumption that the uncertainties referenced can be represented with a random, normal, two-sided probability distribution. This approach has been previously used by Westinghouse for a number of plants, e.g. Wolf Creek.

Instrumentation uncertainties are documented in the Salem RTDP Instrument Uncertainty Methodology Report: Four operating parameter uncertainties are used in the uncertainty analysis of the RTDP. These parameters are pressurizer pressure, primary coolant temperature, reactor power, and RCS flow. Reactor power is monitored by a secondary heat balance once every 24 hours. RCS flow is determined by the performance of a precision flow calorimetric at the beginning of each cycle. The RCS cold leg elbow tap flow indicators are normalized to the precision calorimetric and used for daily RCS flow surveillance. Pressurizer pressure is a control system parameter and the uncertainties associated with that system are included. Similarly, primary coolant temperature, T-average is also a controlled parameter and includes the control system uncertainties.

The RTDP combines error components for an instrument channel by the squareroot sum of the squares (SRSS) method for those uncertainty components found to be independent. Errors that are determined to be dependent are combined arithmetically into independent groups and combined systematically. The described methodology is consistent with previous RTDP submittals and industry standards including ISA S67.04-1982 and staff guidance in Regulatory Guide (RG) 1.105, "Instrument Setpoints," Revision 2 with respect to SRSS, and the guidelines for combining various instrument uncertainties including the relationship between uncertainty components. The licensee stated that Salem-specific instrumentation data and procedures were reviewed and the uncertainty calculations completed based on the use of this data. The calculations are also based on the Salem Units 1 and 2 resistance temperature detector (RTD) bypass elimination design. The staff finds the licensee's approach described above to be consistent with staff guidance.

The staff noted a discrepancy concerning the uncertainty assumptions for primary coolant temperature, T-average in that the RTDP states that only one primary coolant temperature, T_{HOT} RTD is utilized to calculate T-average. The uncertainty calculation itself states that three RTDs are utilized. Since the uncertainty calculation is influenced by the number of RTDs used to calculate the uncertainty term, the licensee has agreed to correct this discrepancy to indicate the calculation utilizes all three RTDs as defined in Table 2 of WCAP-13651 before implementation of the RTDP. Additionally, the cold leg elbow tap flow uncertainty includes additional uncertainties for the elbow tap transmitters. The RTDP utilizes a precision flow calorimetric and generally the cold leg elbow tap transmitter uncertainties are not included based on the normalization of the elbow tap flow instrumentation to the precision flow calorimetric. The Salem uncertainty equations include the additional flow instrumentation uncertainties.

With the incorporation of the RTDP into Salem Units 1 and 2 practices, the licensee has revised the DNB parameters for primary coolant temperature $T_{Average}$, pressurizer pressure, and RCS flow. The revision to TS Table 3.2-1 DNB Parameters is based on the incorporation of RTDP which includes the use of a precision flow calorimetric at the beginning of each cycle to verify the TS DNB reactor coolant system total flow rate parameter and to normalize-the RCS loop flow indicators used for the daily TS RCS flow surveillance. The licensee also plans to revise the Salem Units 1 and 2 FSAR to reflect the incorporation of RTDP as described in WCAP-13651.

Based on the above, the staff finds the methodology chosen by the licensee for margin recovery to be consistent with previously submitted RTDP methodologies, RG 1.105, Rev. 2 and to be compatible with industry accepted standards including ISA S67.04-1982 and is, therefore, acceptable.

2.6 Containment Integrity

The proposed changes to the TSs do not relate to any specific containment system operating limits or surveillance requirements but do affect the containment pressure/temperature response to a LOCA or MSLB. The MRP therefore included new analyses to determine and assure that (1) the maximum peak accident pressure resulting from a LOCA or MSLB will not exceed the containment design pressure, (2) the containment cooling systems are capable of reducing the containment pressure to 50% of the design pressure within 24 hours following a LOCA, and (3) the containment post-DBA temperature profile is bounded by the temperature profile assumption used as a basis for 10 CFR 50.49 qualification of electrical equipment inside containment.

2.6.1 Containment LOCA Response

The licensee's new LOCA containment analyses are described in WCAP-13839, "Fuel Upgrade and Margin Recovery Program: LOCA Containment Integrity Analysis," by J. J. Spryshak and J. A. Kolano, August 1993. The containment LOCA analyses consisted of two portions: (1) an analysis of the mass and energy release from primary and secondary system breaks into containment, and (2) the containment response to the mass and energy release. Bounding initial conditions and conservative assumptions for energy sources and phenomenological processes were assumed, as was a complete spectrum of break sizes and locations and single failures of mitigation systems. The analyses were performed by the vendor using the vendor's NRC-approved thermal-hydraulic analysis codes.

Because approved methods which are applicable to the Salem plant were used, the staff therefore limited the scope of its review to consideration of any changes (from current FSAR analyses) in plant-specific input assumptions that could lead to underprediction of the containment pressure/temperature response.

The Westinghouse standard methodology described in "Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version," WCAP-10325-P-A" was used for the mass and energy release with the exception that steam/water mixing in the broken loop has been credited. This exception involves use of a model based on test data that predicts 100% mixing of steam in the cold leg of the broken loop. The mixing causes condensation of steam that would otherwise be discharged to the containment atmosphere. A staff safety evaluation (C. Rossi to W. Johnson, dated February 17, 1987), approved this change to the 1979 methodology. Based on that evaluation, its use is acceptable for Salem.

No metal-water reaction heat input is assumed in the mass and energy analyses. The SRP Section 6.2.1.3 acceptance criterion for metal-water reaction heat contribution in containment LOCA mass and energy analyses is that it be consistent with the predictions of the Appendix K LOCA peak clad temperature analysis, plus an additional amount be added for conservatism. The licensee has used Appendix K analytical codes as described in the afore-cited approved topical report WCAP-10325, and found that no significant metal-water reaction would occur. Based on use of this conservative methodology, the lack of a metal-water reaction heat contribution in the Salem analysis is acceptable.

The assumed operating power level for the reanalysis was 3479 MWt. This is a reduction from the previous assumption of 3570 MWt. The previous assumption was based on an operating power level greater than that for which the plant is permitted to operate. The licensed power level is 3411 MWt. The use of 3479 MWt, which is 3411 MWt plus 2% power measurement uncertainty, is acceptable.

The new analysis assumes a saturated (rather than superheated) steam generator fluid exit condition. This is conservative for containment peak pressure analyses and is acceptable.

The containment pressure and temperature responses to postulated LOCAs were analyzed using the Westinghouse COCO code described in WCAP-8327, "Containment Pressure Analysis Code (COCO)," July 1974. Changes from the previous COCO model included reduced fan cooler performance (20% reduction), increased safeguards (spray and fan cooler) delay, a 1 degree increase in the RCS temperature uncertainty allowance and reduced safety injection flow. These inputs are more conservative than the previous inputs and are therefore acceptable. The licensee determined that the calculated LOCA maximum peak accident pressure is 41.2 psig and occurs during reflood. The previous value was 45.53 psig. The limiting scenario is a full power, double-ended pump suction break with minimum safeguards (i.e., loss of one train of engineering safety feature). The highest blowdown peak pressure was 39 psig, for a hot leg break. The calculated margin between peak accident pressure and the containment design pressure (47.0 psig) has been increased from 1.47 psi to 5.8 psi. The new peak LOCA pressure "P_a" is bounded by the containment design pressure and is therefore acceptable.

2.6.2 Containment MSLB Response

The licensee's new Main Steam Line Break (MSLB) analyses are described in Section 4.1.18 of Attachment 3 to the application. MSLBs inside containment were analyzed using the LOFTRAN and COCO codes. These are the codes used in previous MSLB analyses. A total of 80 different blowdowns covering four power levels and fourteen break sizes were investigated using the new plant assumptions described above. The results of these analyses indicate that the containment temperature response is within the equipment qualification program limits. The limiting MSLB, 30% power double-ended rupture with feedwater Control Valve Failure, produces a containment peak pressure of approximately 45 psig (from Figure 4.1.18-2 of licensee's submittal), which is greater than that of the limiting LOCA, but less than the containment design pressure.

In a separate licensing action, the licensee submitted a letter dated June 18, 1996, requesting a TS change to the containment design temperature specification, reflecting the new peak MSLB temperature. The staff approved the change in Amendments 198 and 181, dated July 17, 1997.

In conclusion, the licensee analyzed the potential effects of the MRP on the containment responses to primary and secondary pipe breaks. The NRC staff finds that the licensee used conservative analytical methodology and the results are acceptable.

2.7 Radiological Consequences

The parameters and assumptions for the radiological consequence assessments in support of the MRP would be the same as those used in support of the control room envelope modification. The licensee submitted a request for the approval of the control room envelope modification at the Salem Nuclear Generating Station, Unit Nos 1 and 2, with their transmittal letter dated June 10, 1996, and the staff approved the requested modification in Amendment No. 190 for Unit No. 1 and Amendment No. 173 for Unit No. 2, both issued on February 6, 1997. In support of these amendments, the staff performed its independent radiological consequence analyses for the exclusion area boundary, low population zone, and control room operator resulting from postulated design basis accidents. The staff concluded that the radiological consequences were within the dose criteria provided in 10 CFR Part 100, and within the dose criteria specified in General Design Criterion 19 of Appendix A to 10 CFR Part 50. Since the parameters and assumptions are the same, the staff concludes that the radiological consequences for the MRP are acceptable.

2.8 Piping and Supports

By letter dated August 29, 1997, the licensee confirmed that all components of the reactor coolant loop piping and supports meet all licensing basis design requirements and that the operating conditions proposed as part of the MRP are less than the reactor coolant piping design temperature. Thus, the requirements of ASME Section III are satisfied. The staff finds this to be acceptable.

2.9 Proposed Technical Specifications Changes

On the basis of the analysis provided as described above, the licensee has proposed the following changes to the TSs.

e. Table 2.2-1, "Reactor Trip System Instrument Trip Setpoints," was revised to incorporate the change in design RCS flow to 82,500 gpm.

f. Table 2.2-1, Notes 1 and 2, were modified regarding the factors associated with the overtemperature delta temperature calculation and the overpower delta temperature calculation, respectively.

g. TS Bases 2.1.1, "Safety Limits, Reactor Core," was modified to incorporate changes associated with the revised accident and transient analyses and deletion of 3-loop operation requirements.

h. The shutdown margin in TS 3.1.1.1 and SR 4.1.1.1.1 was reduced, and the associated TS Bases B3/4.1.1 and B3/4.1.2 were revised.

i. TS 3.2.2 and TS 3.2.3 were revised to incorporate the safety limit changes for heat flux hot channel factor and the power distribution limits changes for nuclear enthalpy hot channel factor.

j. Table 3.2-1 was changed to incorporate the revised DNB parameters, and the associated TS Bases B3/4.1.1.3 and B3/4.4.1 were revised, accordingly.

g. TS 5.4.2, "Design Features, Volume," incorporated a revised RCS volume and average RCS temperature.

Additionally, the following TS changes were proposed because of the relocation of the unit-specific parameters to the COLR.

a. TS 1.9a and TS 6.9.1.9 were added to incorporate the requirement for a unit-specific document that provides the core operating limits for the current operating reload cycle.

b. References to parameters in COLR were incorporated into SR 4.1.1.1.1, TS 3.1.1.3, SR 4.1.1.3, TS 3.1.3.1, TS 3.1.3.5, Fig. 3.1-1(removed), TS 3.2.1, SR 4.2.1.1, Fig. 3.2-1, SR 4.2.2.2, Fig 3.2-2, and TS Bases B3/4.2.1, B3/4.2.2 and B3/4.2.3

The discussion of 3-loop operation was deleted since it was not approved by the NRC. Accordingly, TS 2.1.1 was modified and references to future Figures 2.1-2 and 3.1-2 were removed.

The NRC staff finds the proposed changes to the TSs consistent with the licensee's revised analyses. The relocation of unit-specific parameters to the COLR was consistent with the guidance in NRC Generic Letter 88-16. The NRC staff also found these changes and the other editorial changes acceptable.

In addition, as an administrative matter, the staff corrected minor clerical error in TS 3.1.3.3 and TS 3.1.3.5 to make them consistent with similar TSs issued for Salem Unit 1.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (61 FR 34898). The amendment also changes reporting or recordkeeping requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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