May 9, 1989

Docket Nos. 50-272/311

Mr. Steven E. Miltenberger
Vice President and Chief Nuclear
Officer
Public Service Electric & Gas Company
Post Office Box 236
Hancocks Bridge, New Jersey 08038

Dear Mr. Miltenberger:

SUBJECT: USE OF VANTAGE 5 HYBRID FUEL (TAC NOS. 71836/71837)

RE: SALEM GENERATING STATION, UNIT NOS. 1 AND 2

The Commission has issued the enclosed Amendment Nos. 96 and 72 to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Generating Station, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated December 30, 1988 and supplemented by letters dated April 19 and May 4, 1989, which provided corrected technical specification pages which did not change the technical requirements and a commitment to revise FSAR Section 15.4.5.

These amendments allow the use of Vantage 5 Hybrid Fuel; reduce the flow measurement uncertainty allowance and eliminate the rod bow penalty.

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/

James C. Stone, Project Manager
Project Directorate I-2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 96 to
   License No. DPR-70
2. Amendment No. 72 to
   License No. DPR-75
3. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION w/enclosures:

Docket File	MO'Brien (2)	Wanda Jones	Brent Clayton	NRC PDR
OGC	EButcher	EWenzinger	Local PDR	DHagan
WHodges, SRXB	PDI-2 Reading	EJordan	ACRS (10)	WButler
BGrimes	CMiles, GPA/PA	JStone/MThadani		SLWu
TMeek (8)	RDiggs, ARM/LFMB			

WB [MILT LET]

Previously concurred*

PDI-2/LA*	PDI-2/PM*	OGC*	MCThadani for PDI-2/D*
MO'Brien	JStone:tr	MYoung	WButler
05/01/89	05/01/89	05/05/89	5/8/89

CP-1
DF01
1/1

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

May 9, 1989

Docket Nos. 50-272/311

Mr. Steven E. Miltenberger
Vice President and Chief Nuclear Officer
Public Service Electric & Gas Company
Post Office Box 236
Hancocks Bridge, New Jersey 08038

Dear Mr. Miltenberger:

SUBJECT: USE OF VANTAGE 5 HYBRID FUEL (TAC NOS. 71836/71837)

RE: SALEM GENERATING STATION, UNIT NOS. 1 AND 2

The Commission has issued the enclosed Amendment Nos. 96 and 72 to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Generating Station, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated December 30, 1988 and supplemented by letters dated April 19 and May 4, 1989, which provided corrected technical specification pages which did not change the technical requirements and a commitment to revise FSAR Section 15.4.5.

These amendments allow the use of Vantage 5 Hybrid Fuel; reduce the flow measurement uncertainty allowance and eliminate the rod bow penalty.

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

Sincerely,

James C. Stone

James C. Stone, Project Manager
Project Directorate I-2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 96 to License No. DPR-70
2. Amendment No. 72 to License No. DPR-75
3. Safety Evaluation

cc w/enclosures:
See next page

Mr. Steven E. Miltenberger
Public Service Electric & Gas Company

Salem Nuclear Generating Station

cc:

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Wilmington, DE 19899

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

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-272

SALEM GENERATING STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 96
License No. DPR-70

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 December 30, 1988 and supplemented by letters dated April 19 and May 4, 1989, 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-70 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. 96 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

/s/

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 9, 1989

PDI-2/LA
MO'Brien
5/1/89

PDI-2/PM
JStone
5/1/89

OGC
MYoung
5/5/89

PDI-2/D
WButler
5/8/89

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 96 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 9, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 96

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Revise Appendix A as follows:

Remove Pages	Insert Pages
B 2-1	B 2-1
B 2-4	B 2-4
B 2-6	B 2-6
3/4 1-21	3/4 1-21
3/4 2-9	3/4 2-9
3/4 2-14	3/4 2-14
B 3/4 2-6	B 3/4 2-6

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 the W-3, R-Grid correlation for standard (LOPAR) fuel assemblies and WRB-1 correlation for Vantage 5H fuel assemblies. The DNB correlation has 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, defined 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: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 or W-3 R-Grid correlation). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR correlation limit (1.17 for the WRB-1 or 1.30 for the W-3 R-Grid).

The curves of Figures 2.1-1 and 2.1-2 show 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.

BASES

The Power Range Negative Rate trip provides protection to ensure that the minimum DNBR is maintained above the design DNBR value for multiple control rod drop accidents. The analysis of a single control rod drop accident indicates a return to full power may be initiated by the automatic control system in response to a continued full power turbine load demand or by the negative moderator temperature feedback. This transient will not result in a DNBR of less than the design DNBR value, therefore single rod drop protection is not required.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This 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. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

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.

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 and anticipated transients when 3 loops are in operation and the Overtemperature ΔT trip set point is adjusted to the value specified for all loops in operation. With the Overtemperature ΔT trip set point adjusted to the value specified for 3 loop operation, the P-8 trip at 76% RATED THERMAL POWER will prevent the minimum value of the DNBR from going below the design DNBR value during normal operational transients and anticipated transients with 3 loops in operation.

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.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall

REACTIVITY CONTROL SYSTEMS

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.3 The individual full length (shutdown and control) rod drop time from 228 steps withdrawn shall be ≤ 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} ≥ 541°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODE 3.

ACTION:

a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with 3 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to ≤71% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.3 The rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

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

LIMITING CONDITION FOR OPERATION

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

$$F^N_{\Delta H} \leq 1.55\ [1.0+0.3(1-P)]$$

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

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	3 Loops In Operation
Reactor Coolant System T_{avg}	≤ 582°F	≤ 572°F
Pressurizer Pressure	≥ 2220 psia*	≥ 2220 psia*
Reactor Coolant System	≥ 357,200 gpm#	≥ 284,500 gpm#

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

#Includes a 2.2% flow measurement uncertainty plus a 0.1% measurement uncertainty due to feedwater venturi fouling.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of the design DNBR value throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters thru instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

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

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-311

SALEM GENERATING STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.72
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 December 30, 1988 and supplemented by letters dated April 19 and May 4, 1989, 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) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 72 , 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 fuel load during the fifth refueling outage currently scheduled to begin in March 1990.

FOR THE NUCLEAR REGULATORY COMMISSION

/s/
Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 9, 1989

PDI-2/LA	PDI-2/PM	OGC	PDI-2/D
MO'Brien	JStone:tr		WButler
5/1/89	5/1/89	5/5/89	5/8/89

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 72 , 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 fuel load during the fifth refueling outage currently scheduled to begin in March 1990.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 9, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 72

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Revise Appendix A as follows:

Remove Pages	Insert Pages
B 2-1	B 2-1
B 2-4	B 2-4
B 2-6	B 2-6
3/4 1-18	3/4 1-18
3/4 2-9	3/4 2-9
3/4 2-10	3/4 2-10
3/4 2-16	3/4 2-16
3/4 2-17	3/4 2-17
B 3/4 2-4	B 3/4 2-4
B 3/4 2-5	B 3/4 2-5
B 3/4 2-6	B 3/4 2-6

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 the W-3, R-Grid correlation for standard (LOPAR) fuel assemblies and WRB-1 correlations for Vantage 5H fuel assemblies. The DNB correlation has 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, defined 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: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 or W-3, R-Grid correlation). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR correlation limit (1.17 for the WRB-1 or 1.30 for the W-3 R-Grid).

The curves of Figures 2.1-1 and 2.1-2 show 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, $F^N_{\Delta H}$ of 1.55 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:

$$F^N_{\Delta H} \leq 1.55\ [1.0+0.3(1-P)]$$

where P is the fraction of RATED THERMAL POWER

These limiting heat flux conditions are higher than those calculated for the range of all control rods 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

LIMITING SAFETY SYSTEM SETTINGS

BASES

The Power Range Negative Rate trip provides protection to ensure that the minimum DNBR is maintained above the design DNBR value for control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which, when in conjunction with nuclear power being maintained equivalent to turbine power by action of the automatic rod control system, could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor for all single or multiple dropped rods.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature Delta T

The Overtemperature delta T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This 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. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

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 and anticipated transients when 3 loops are in operation and the Overtemperature delta T trip set point is adjusted to the value specified for all loops in operation. With the Overtemperature delta T trip set point adjusted to the value specified for 3 loop operation, the P-8 trip at 76% RATED THERMAL POWER will prevent the minimum value of the DNBR from going below the design DNBR value during normal operational transients and anticipated transients with 3 loops in operation.

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.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability of the Reactor Protection System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by greater than or equal to 1.42×10^6 lbs/hour. The Steam Generator Low Water level portion of the trip is activated when the water level drops below 24 percent, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

REACTIVITY CONTROL SYSTEMS

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.3 The individual full length (shutdown and control) rod drop time from 228 steps withdrawn shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 541°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 & 2.

ACTION:

a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with 3 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 76% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.3 The rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

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:

$$F^N_{\Delta H} \leq 1.55\ [1.0 + 0.3\ (1.0-P)]$$

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

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 $\leq$ 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.

SURVEILLANCE REQUIREMENTS

4.2.3.1 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map:

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

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

c. The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The measured $F^N_{\Delta H}$ of 4.2.3.1 above, shall be increased by 4% for measurement uncertainty.

POWER DISTRIBUTION LIMITS

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}.

b. Pressurizer Pressure.

c. Reactor Coolant System Total Flow Rate.

APPLICABILITY: MODE 1

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System Total Flow Rate shall be determined to be within its limit by measurement at least once per 18 months.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 4 Loops in Operation
Reactor Coolant System T_{avg}	≤ 582°F
Pressurizer Pressure	≤ 2220 psia*
Reactor Coolant System	≥ 357200 gpm#

*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 2.2% flow uncertainty plus a 0.1% measurement uncertainty due to feedwater venturi fouling.

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 by more than ± 12 steps from the group demand position.

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

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

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

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

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 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 $F^N_{\Delta H} \leq 1.55/1.08$. 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. abormal 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, FQ(Z), remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F^{RTP}_{xy}), as provided in the Radial Peaking Factor Limit Report per specification 6.9.1.10, was determined from expected power control maneuvers over the full range of burnup conditions in the core.

3/4.2.4 QUADRANT POWER TILT RATIO

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

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

The 2 hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3% from RATED THERMAL POWER for each percent of tilt in excess of 1.0.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained with the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of the design DNBR value throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 96 AND 72 TO FACILITY OPERATING

LICENSE NOS. DPR-70 AND DPR-75

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

SALEM GENERATING STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-272 AND 50-311

1.0 INTRODUCTION

By letter dated December 30, 1988, and supplemented by letters dated April 19 and May 4, 1989, Public Service Electric & Gas Company requested an amendment to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Generating Station, Unit Nos. 1 and 2. The supplemental letters provided corrected technical specification pages which did not change the technical requirements and a commitment to revise FSAR Section 15.4.5. The supplemental letters did not affect the action as noticed in the Federal Register or alter the staff's initial determination.

The licensee's request was for the Technical Specification changes resulting from the use of VANTAGE 5 Hybrid (VANTAGE 5H) fuel assemblies for Salem Unit 1 Cycle 9 and Unit 2 Cycle 6 reload core and future cores. The VANTAGE 5H fuel design evolves from the VANTAGE 5, Optimized Fuel Assembly (OFA), and Standard (STD) fuel assembly designs. The features of the VANTAGE 5H fuel assembly consist of reconstitutable top nozzles, Zircaloy grids, Debris Filter Bottom Nozzles (DFBNs), and the capability of achieving high burnups. These features were previously reviewed and approved by NRC in the Westinghouse topical report WCAP-10444-P-A, "Referenced Core Report VANTAGE 5 Fuel Assembly" Addendum 2.

The licensee also plans to remove thimble plugging devices from the Salem cores. Thimble plugging devices are used in Salem Units to limit the bypass flow. These fuel assembly guide thimble tubes that are not in RCCA locations or are not equipped with sources or burnable absorbers have thimble plugs inserted in them.

During the review of VANTAGE 5 fuel design in WCAP-10444-P-A, we identified a few conditions to be resolved for those licensees using VANTAGE 5 fuel design. Since the VANTAGE 5H fuel design adopts some features from the VANTAGE 5 fuel design, our review and evaluation will address those conditions listed in the safety evaluation of WCAP-10444-P-A that affect Salem's VANTAGE 5H fuel.

2.0 EVALUATION

(1) Statistical Convolution Method

In our SER on WCAP-10444, we stated that the statistical convolution method should not be used in VANTAGE 5 for evaluating the fuel rod shoulder gap. The licensee indicated that the statistical convolution method was not used for the VANTAGE 5H fuel design and the currently approved method was used for evaluating fuel rod shoulder gap. Therefore we consider this acceptable.

(2) Irradiation Demonstration Program

In our SER on WCAP-10444, we required that an irradiation program be performed to confirm the VANTAGE 5 fuel performance. The licensee stated that there were numerous demonstration programs involving OFA fuel assemblies containing Zircaloy grids irradiated in 14X14, 15X15 and 17X17 cores. The satisfactory performance of these demonstration assemblies resulted in OFA with Zircaloy grids in reload applications in many Westinghouse reactors. The OFA fuel assemblies with Zircaloy grids cover the VANTAGE 5H fuel design features; we thus conclude that the VANTAGE 5H fuel assemblies will perform satisfactorily in Salem.

(3) Improved Thermal Design Procedure (ITDP)

In our SER on WCAP-10444, we stated that those restrictions in approving the use of Westinghouse improved thermal design procedure (ITDP) should be applied to the VANTAGE 5 fuel design. The licensee indicated that they conformed to these restrictions of ITDP for Salem. We therefore conclude that this is acceptable.

(4) DNBR Limit

In our SER on WCAP-10444, we stated that plant-specific analysis should be performed to show that the DNBR limit is not violated with the higher value of $F^N_{\Delta H}$. The licensee indicated that the VANTAGE 5H fuel does not employ higher values of $F^N_{\Delta H}$ and F_Q thus no reanalysis of DNBR transients is needed. We therefore consider that this condition is satisfied for VANTAGE 5H fuel in Salem.

(5) Positive Moderator Temperature Coefficient (MTC)

In our SER on WCAP-10444, we stated that if a positive moderator temperature coefficient (MTC) is intended, the same positive MTC should be used in the plant specific analysis. The licensee indicated that no positive MTC was considered in the submittal. We thus consider that this condition is satisfied for VANTAGE 5H fuel in Salem.

(6) Reactor Coolant Pump Shaft Seizure

In our SER on WCAP-10444, we stated that mechanistic approach (2700°F peak clad temperature) in determining the fraction of fuel failures during the reactor coolant pump seizure accident was unacceptable; the fuel failure criterion should be the 95/95 DNBR limit. The licensee reanalyzed the reactor coolant pump shaft (locked rotor) accident based on a failure criterion of peak clad temperature of 2700°F. The licensee concluded that there was no fuel failure and the coolability was maintained since the calculated peak clad temperature (2043°F) remained much less than 2700°F and the amount of Zirconium-water reaction was small. As indicated above, we disapprove the use of mechanistic approach based on 2700°F peak clad temperature in determining the fuel failure. We require the licensee to modify the fuel failure criterion based on the approved 95/95 DNBR limit. However, since the VANTAGE 5H fuel does not employ higher values of $F^N_{\Delta H}$ and F_Q, we believe that there is no need to reanalyze this accident based on the 95/95 DNBR fuel failure criterion before operation of Salem with VANTAGE 5H fuel.

By letter dated May 4, 1989, the licensee committed to modify the fuel failure criterion described in FSAR Section 15.4.5 so that it is based on the 95/95 DNBR limit. The next FSAR update will be mid 1990. The staff finds this acceptable.

TECHNICAL SPECIFICATION CHANGES

The proposed Technical Specification changes are related to the use of VANTAGE 5H fuel, a new DNBR correlation, and a new rod bow penalty methodology. Salem Unit 2 Technical Specifications have some variations from the Unit 1. We discuss these changes in the following:

(1) Section 2.1.1 Basis, Pages B2-1, B2-4 and B2-6 for Units 1 and 2

The old W-3 DNBR correlation is changed to W-3 (R-Grid) correlation for Standard fuel design. A new DNBR correlation of WRB-1 is added for VANTAGE 5H fuel design. All these DNBR correlations are approved for use in licensing applications. We thus consider these changes acceptable.

(2) Section 3/4.2.5 Basis, Page B3/4 2-6 for Units 1 and 2

The old phrase of minimum DNBR limit is changed to the design DNBR value because there are two different DNBR correlations intended for two different fuel designs. We consider this change acceptable.

(3) Section 3.1.3.3, Page 3/4 1-21 for Units 1 and Page 3/4 1-18 for Unit 2

The rod drop time is revised to be less than or equal to 2.7 seconds due to the use of VANTAGE 5H fuel design. The licensee has taken into account the effect of the increased rod drop time in all safety analyses. We thus consider this change acceptable.

(4) Section 3.2.3, Pages 3/4 2-9, 3/4 2-10a and 3/4 2-14 for Unit 1

The rod bow penalty is revised to incorporate a new methodology which reduces the rod bow penalty. The licensee has demonstrated that the use of new DNBR correlations has enough margin to offset the rod bow penalty at burnups greater than 24,000 MWd/MTU. We thus consider that the reduced rod bow penalty is acceptable for Unit 1.

(5) Section 3.2.3 Page 3/4 2-9, Section 4.2.3.2 Pages 3/4 2-10, 3/4 2-16 and 3/4 2-17, and Section 3/4.2.3 Basis Pages B3/4 2-4 and B3/4 2-5 for Unit 2

The revisions in these pages for Unit 2 are for the revised rod bow penalty as discussed above for Unit 1. We thus consider these changes acceptable for Unit 2.

THIMBLE PLUG REMOVAL

In order to limit the core bypass flow, there were thimble plug devices inserted in those guide thimble tubes which were not under RCC locations or were not equipped with sources and burnable absorbers. The devices resulted in a net gain of about 2 percent in DNBR margin. The licensee intends to remove all the thimble plug devices from the core during the transition core and all VANTAGE 5H fueled core.

The licensee analyzed the impact of thimble plug removal based on the mechanical and thermal-hydraulic design consideration. The licensee concluded that the major result of thimble plug removal is the increase of core bypass flow. The increasing bypass flow has been considered in the non-LOCA and LOCA safety analyses and no adverse impact was discovered. Based on the licensee favorable results, we conclude that the thimble plug removal is adequately addressed and the adverse impact is minimal.

We have reviewed the licensee's submittal of VANTAGE 5H fuel design and Technical Specification changes for Salem Units 1 and 2 transition cores and all VANTAGE 5H cores. Based on the approved generic topical report WCAP-10444-P-A and plant-specific analyses, we approve the use of VANTAGE 5H fuel design and Technical Specification changes for Salem. The commitment to modify the fuel failure criterion in FSAR Section 15.4.5, so that it is based on the 95/95 DNBR limit, in the 1990 FSAR update is acceptable.

3.0 COMMENTS

The State of New Jersey by letter dated March 6, 1989, commented that VANTAGE 5H fuel uses grids of Zircalloy-4 instead of Inconel. The concern was that the significant hazards consideration provided by the licensee did not include an evaluation of the potential generation of an excessive amount of hydrogen as a result of metal-water reaction involving the grids and the reactor coolant under a postulated Loss of Coolant Accident (LOCA). The State of New Jersey felt the potential for excessive hydrogen generation warranted an evaluation before a determination of no significant hazard consideration could be rendered.

The NRC responded to the State's concern by letter dated May 2, 1989. That letter concluded that the use of Zircalloy in lieu of Inconel for the fuel element grid would involve no significant hazards consideration based on:

1. The VANTAGE 5H fuel design has been reviewed and approved by the NRC on a generic basis, without conditions pertaining to the use of Zircalloy in core structural components.

2. The melting temperature of Zircalloy (3362°F) is much higher than Inconel (2346°F), therefore the fuel bundles will be maintained in a coolable geometry for a longer period of time.

3. The weight of Zircalloy in the grids is a small fraction of the total Zircalloy in the core which reduces the affect of any metal-water reaction involving the grids on the overall hydrogen generation.

4. The use of Zircalloy for core structural components has been previously approved in earlier core designs.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The staff has determined that the amendments involve 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 amendments involve no significant hazards consideration and the State comments on such findings have been addressed. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (54 FR 6207) on February 8, 1989 and consulted with the State of New Jersey. No public comments other than from the State were received.

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security to the health and safety of the public.

Principal Contributor: S. L. Wu

Dated: May 9, 1989