Csp 11/1
Docket File

October 30, 1979

Docket No. 50-272

Mr. F. P. Librizzi, General Manager
Electric Production
Public Service Electric and Gas Company
80 Park Place, Room 7221
Newark, New Jersey 07101

REGULATORY DOCKET FILE COPY

Dear Mr. Librizzi:

The Commission has issued the enclosed Amendment No. 20 to Facility Operating License No. DPR-70 for the Salem Nuclear Generating Station, Unit No. 1. This amendment consists of changes to the Technical Specifications in response to your request dated March 2, 1979 with additional information dated April 30, 1979, July 25, 1979, August 3, 1979, August 8, 1979, August 9, 1979, and September 14, 1979.

The amendment revises Radiological Safety Technical Specifications related to the Cycle 2 core. These changes involve the axial flux difference, heat flux hot channel factor, nuclear enthalpy hot channel factor and reactor core. This amendment authorizes initiation of operation for Cycle 2.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed. In addition to the revisions in the Technical Specifications, our Safety Evaluation also addresses potential safety questions identified in Bulletins issued by the NRC Office of Inspection and Enforcement as well as in Licensee Event Reports issued from your office during the reload outage.

We have also performed a preliminary review of your October 4, 1979 response to our September 14, 1979 letter regarding potential malfunctions due to high energy line breaks affecting safety-related systems. Although

Cp 2
CP

OFFICE ➤					7911080135	
SURNAME ➤						
DATE ➤						

our review is continuing, we have determined that there is no need to modify, suspend, or revoke your license at this time due to this concern.

We have also made a preliminary biological assessment of the potential damage to short-nose sturgeon, an endangered species of fish, resulting from renewed operation of Unit No. 1. On the basis of this assessment we have determined that continued operation will not result in significant impact on the population of this fish in the Delaware River. Likewise, as required by the Endangered Species Act, as amended, we have found that renewed operation "will not foreclose formulation or implementation of any reasonable and prudent alternative measures which would avoid jeopardizing the continued existence of this endangered species or adversely modifying or destroying the critical habitat of any such species." In further compliance with the Endangered Species Act, we are initiating formal consultation with other responsible regulatory agencies relative to this potential problem. Should this consultation indicate that further actions should be taken, we shall advise you in a timely manner.

Sincerely,

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. to DPR-70
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
See next page

OFFICE	DOR:ORB1 AS					
SURNAME	ASchwencer:jb					
DATE	10/30/79					

Mr. F. P. Librizzi
Public Service Electric and Gas Company - 2 -

our review is continuing, we have determined that there is no need to modify, suspend, or revoke your license at this time due to this concern.

Sincerely,

Original Signed By

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 28 to DPR-70
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
See next page

Distribution

Docket File 50-272	I&E (5)
NRC PDR	B. Jones (4)
Local PDR	B. Scharf (10)
NRR Rdg	D. Brinkman
ORB1 Rdg	B. Harless
D. Eisenhut	C. Miles
B. Grimes	R. Diggs
W. Gammill	H. Denton
T. J. Carter	ACRS (16)
C. Parrish	TERA
W. Ross	J. Buchanan
Attorney, OELD	

AS 10/30/79 with corrections & additions re fish kill

AS 10/25/79

TAC 11543

#8
2981-29

OFFICE	DOR:ORB1	DOR:ORB1	DOR:ORB1	DOR:AD:ORP	OELD
SURNAME	WJRoss:jb	CSParrish	ASchwencer	WPGammill	BHB
DATE	10/19/79	10/23/79	10/25/79	10/ /79	10/24/79

NRC FORM 318 (9-76) NRCM 0240 ★ U.S. GOVERNMENT PRINTING OFFICE: 1978 - 265 - 769

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

October 30, 1979

Docket No. 50-272

Mr. F. P. Librizzi, General Manager
Electric Production
Public Service Electric and Gas Company
80 Park Place, Room 7221
Newark, New Jersey 07101

Dear Mr. Librizzi:

The Commission has issued the enclosed Amendment No. 20 to Facility Operating License No. DPR-70 for the Salem Nuclear Generating Station, Unit No. 1. This amendment consists of changes to the Technical Specifications in response to your request dated March 2, 1979 with additional information dated April 30, 1979, July 25, 1979, August 3, 1979, August 8, 1979, August 9, 1979, and September 14, 1979.

The amendment revises Radiological Safety Technical Specifications related to the Cycle 2 core. These changes involve the axial flux difference, heat flux hot channel factor, nuclear enthalpy hot channel factor and reactor core. This amendment authorizes initiation of operation for Cycle 2.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed. In addition to the revisions in the Technical Specifications, our Safety Evaluation also addresses potential safety questions identified in Bulletins issued by the NRC Office of Inspection and Enforcement as well as in Licensee Event Reports issued from your office during the reload outage.

We have also performed a preliminary review of your October 4, 1979 response to our September 14, 1979 letter regarding potential malfunctions due to high energy line breaks affecting safety-related systems. Although our review is continuing, we have determined that there is no need to modify, suspend, or revoke your license at this time due to this concern.

We have also made a preliminary biological assessment of the potential damage to short-nose sturgeon, an endangered species of fish, resulting from renewed operation of Unit No. 1. On the basis of this assessment we have determined that continued operation will not result in significant impact on the population of this fish in the Delaware River. Likewise, as required by the Endangered Species Act, as amended, we have found that renewed operation "will not foreclose formulation or implementation of any reasonable and prudent alternative measures which would avoid jeopardizing the continued existence of this endangered species or adversely modifying or destroying the critical habitat of any such species." In further compliance with the Endangered Species Act, we are initiating formal consultation with other responsible regulatory agencies relative to this potential problem. Should this consultation indicate that further actions should be taken, we shall advise you in a timely manner.

Sincerely,

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 20 to DPR-70
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
See next page

cc: Mark J. Wetterhahn, Esquire
Conner, Moore and Corber
Suite 1050
1747 Pennsylvania Avenue, NW
Washington, D. C. 20006

Richard Fryling, Jr., Esquire
Assistant General Solicitor
Public Service Electric and Gas Company
80 Park Place
Newark, New Jersey 07101

Gene Fisher,.Bureau of Chief
Bureau of Radiation Protection
380 Scotch Road
Trenton, New Jersey 08628

Mr. Hank Midura, Manager
Salem Nuclear Generating Station
Public Service Electric and Gas Company
80 Park Place
Newark, New Jersey 07101

Salem Free Library
112 West Broadway
Salem, New Jersey 08079

Leif J. Norrholm
U. S. Nuclear Regulatory Commission
Drawer I
Hancocks Bridge, New Jersey 08038

Attorney General
Department of Law and Public Safety
State House Annex
Trenton, New Jersey 08625

Samuel E. Donelson, Mayor
Lower Alloways Creek Township
Municipal Hall
Hancocks Bridge, New Jersey 08038

Richard B. McGlynn, Commissioner
Department of Public Utilities
State of New Jersey
101 Commerce Street
Newark, New Jersey 07102

Deputy Attorney General
State House Annex
State of New Jersey
36 West State Street
Trenton, New Jersey 08625

Director, Technical Assessment Divisio
Office of Radiation Programs (AW-459)
U. S. Environmental Protection Agency
Crystal Mall #2
Arlington, Virginia 20460

U. S. Environmental Protection Agency
Region II Office
ATTN: EIS COORDINATOR
26 Federal Plaza
New York, New York 10007

Mr. E. N. Schwalje, Manager
of Quality Assurance
Public Service Electric and Gas Compar
80 Park Place
Newark, New Jersey 07101

Mr. R. L. Mittl, General Manager
Licensing and Environment
Public Service Electric and Gas Compar
80 Park Place
Newark, New Jersey 07101

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

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
PHILADELPHIA ELECTRIC COMPANY
DELMARVA POWER AND LIGHT COMPANY
ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-272

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 20
License No. DPR-70

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

A. The application for amendment by Public Service Electric and Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) dated March 2, 1979, as revised by letters dated April 30, 1979, July 25, 1979, August 3, 1979, August 8, 1979, August 9, 1979, and September 14, 1979 for potential safety questions 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;

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.

7911080137

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

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 20 , 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 30, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 20

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-7	3/4 2-7
	3/4 2-7a
3/4 2-8	3/4 2-8
3/4 2-9	3/4-2-9
	3/4 2-10a
B 3/4 2-1	B 3/4 2-1
B 3/4 2-4	B 3/4 2-4
B 3/4 2-5	B 3/4 2-5

3/4.2 POWER DISTRIBUTION LIMITS

AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within a ±5% target band (flux difference units) about the target flux difference. In addition, during the first 72 EFPD (2700 MWD/MTU) operation in Cycle 2, the indicated AFD shall be maintained less than +7.5% at RATED THERMAL POWER with the allowed AFD increasing by 1.0% for each 1.0% reduction in THERMAL POWER.

APPLICABILITY: MODE 1 ABOVE 50% RATED THERMAL POWER*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the above limits 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 above limits for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

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

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

*See Special Test Exception 3.10.2

LIMITING CONDITION FOR OPERATION (Continued)

b. THERMAL POWER shall not be increased above 90% of RATED THERMAL POWER unless the indicated AFD is within the above limits 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 above limits for more than 1 hour penalty deviation cumulative during the previous 24 hours.

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

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

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

b. One-half minute penalty deviation for each one minute of POWER OPERATION outside of the limits at THERMAL POWER levels below 50% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

1. For all core planes containing bank "D" control rods;
   $F_{xy}^{RTP} \leq 1.92$ for core elevations up to 6.0 ft.,
   $F_{xy}^{RTP} \leq 1.89$ for core elevations from 6.0 to 12.0 ft., and

2. For all unrodded planes;
   $F_{xy}^{RTP} \leq 1.67$ for core elevations up to 6.0 ft., and
   $F_{xy}^{RTP} \leq 1.65$ for core elevations from 6.0 to 12.0 ft.

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_{xy}^{C} exceeds F_{xy}^{L}.

SURVEILLANCE REQUIREMENTS (Continued)

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.

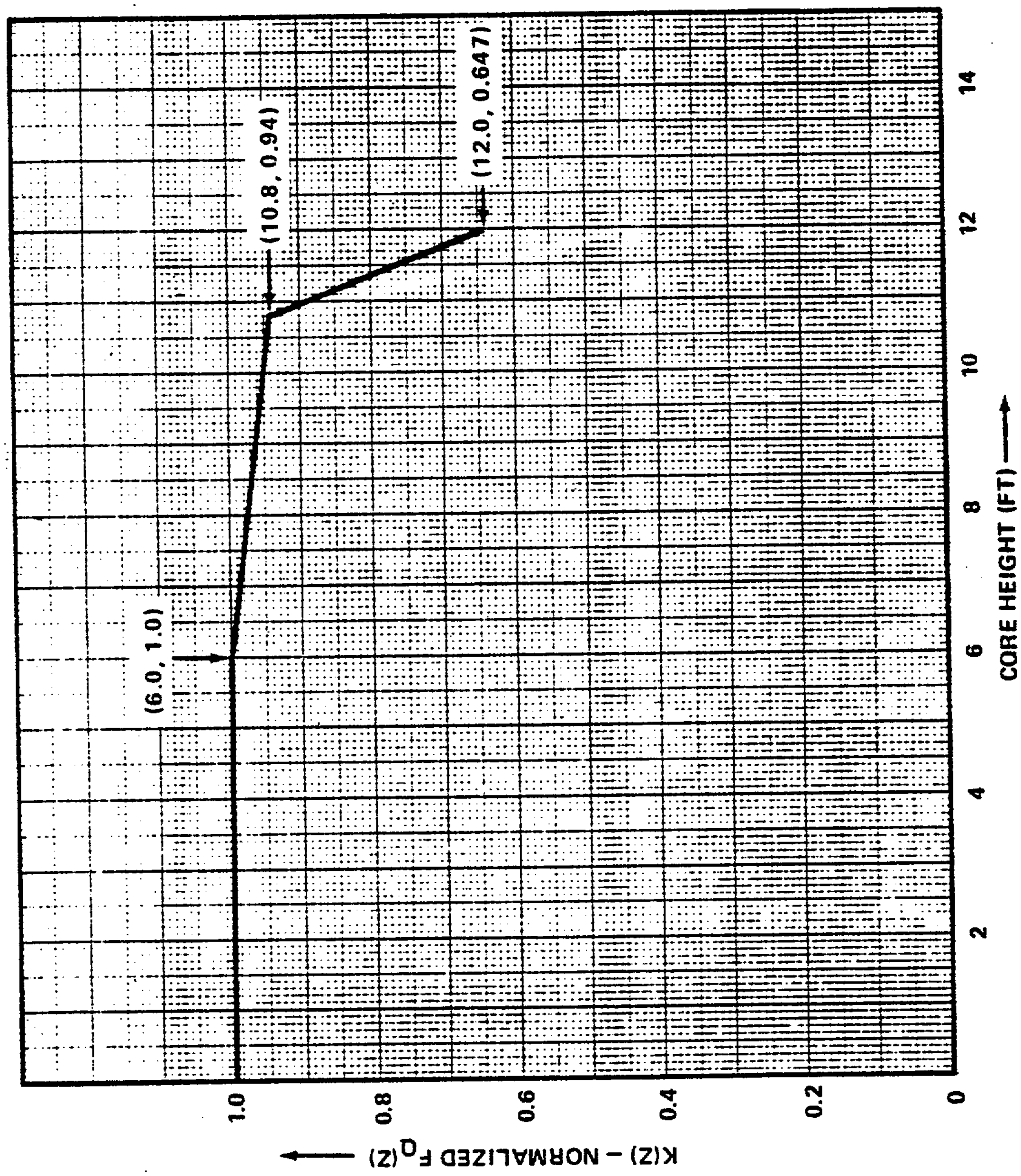

FIGURE 3.2-2

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

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

LIMITING CONDITION FOR OPERATION

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

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

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

RBP(BU)= Rod Bow Penalty as a function of region average burnup as shown in Figure 3.2-3, where a region is defined as those assemblies with the same loading date (reloads) or enrichment (first core).

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.

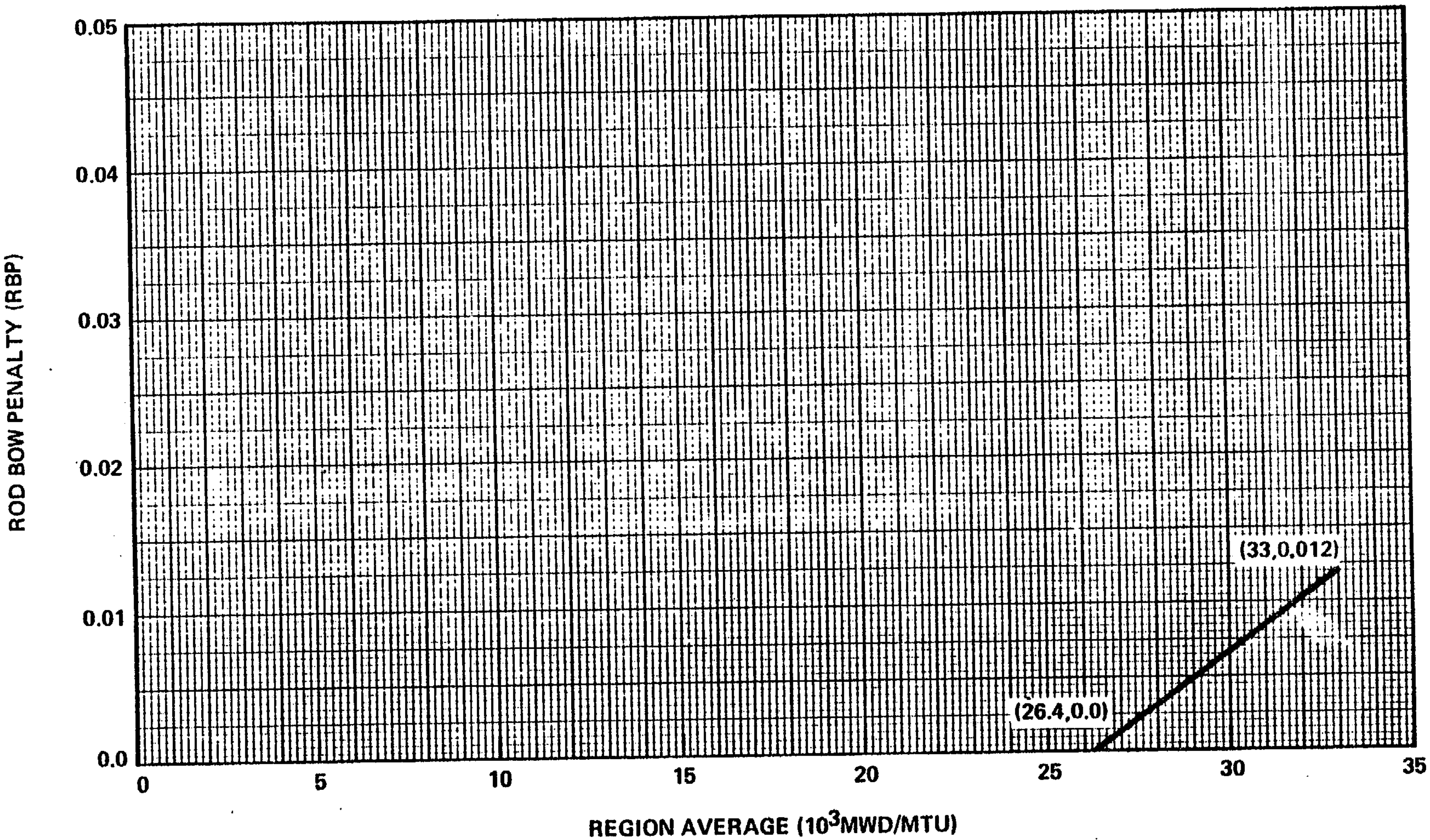

Figure 3.2-3
ROD BOW PENALTY AS A FUNCTION OF BURNUP

BASES

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

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

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

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

$F_{xy}(Z)$ 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 2.32 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. 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.

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the $\pm 5\%$ target band 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 of Figure 3.2-1 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. A first alarm is received any time the auctioneered high flux difference exceeds the target band limits. A second alarm is received if the AFD exceeds its allowable limits for a cumulative time of one hour during any 24 hour time period starting with the occurrence of the first alarm. Time outside the target band is graphically presented on the strip chart.

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

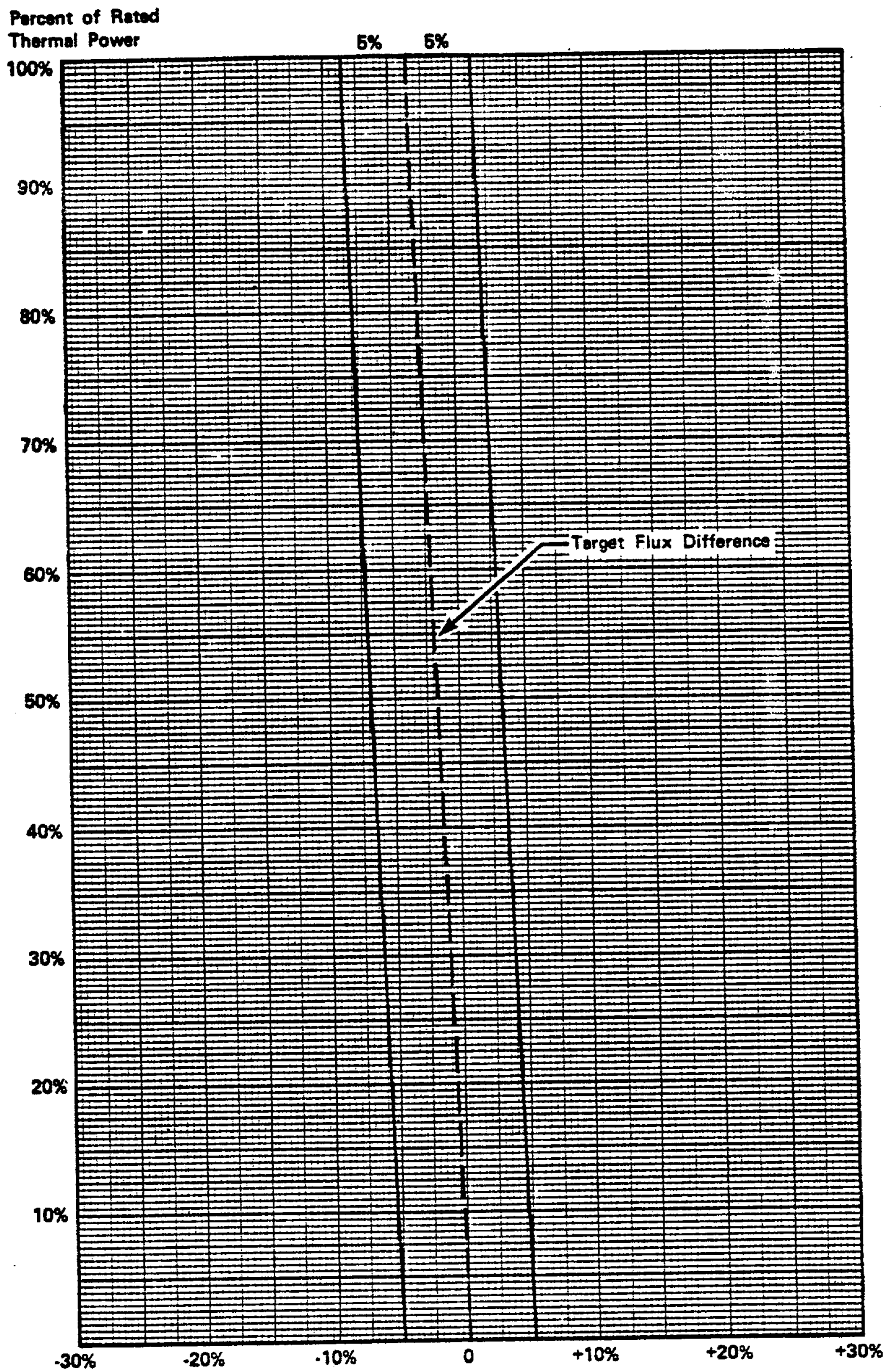

Figure B 3/4 2-1 TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL AND RADIAL PEAKING FACTORS-

$F_Q(Z)$, $F^N_{\Delta H}$ and $F_{xy}(Z)$

The limits on heat flux and nuclear enthalpy hot channel factors ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these hot channel factors are measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the hot channel factor limits are maintained provided:

a. Control rod in a single group move together with no individual rod insertion differing by more than ± 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. 5% is the appropriate allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

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

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

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

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

The radial peaking factor, F_{xy} (Z), is measured periodically to provide additional assurance that the hot channel factor, F_Q (Z), remains within its limits. The F_{xy} (Z) limits were 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 two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent from RATED THERMAL POWER for each percent of tilt in excess of 1.0.

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 1.30 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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 20 TO FACILITY OPERATING LICENSE NO. DPR-70

PUBLIC SERVICE ELECTRIC AND GAS COMPANY,
PHILADELPHIA ELECTRIC COMPANY,
DELMARVA POWER AND LIGHT COMPANY, AND
ATLANTIC CITY ELECTRIC COMPANY

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

DOCKET NO. 50-272

Introduction

By letter dated March 2, 1979 (Reference 1) and supplemented by letters dated April 30, 1979 (Reference 2), August 8, 1979 (Reference 3), and August 9, 1979 (Reference 4), Public Service Electric and Gas Company (the licensee) proposed changes to the Technical Specifications for Salem Generating Station Unit 1 for Cycle 2 operation. This document represents the NRC staff's evaluation of these proposals.

During the refueling outage the licensee notified the NRC that fuel assembly grid strap damage (Reference 5) and broken RCCA rodlets (Reference 6) had been observed in the Cycle 1 core. The effect of these structural problems on the Cycle 2 core has been reviewed as part of our reload evaluation.

The refueling outage for Salem Unit No. 1 also coincided with a period when the staff was reviewing the safety of all operating nuclear power plants in light of several potential safety problems identified by the NRC Office of Inspection and Enforcement (IE). These concerns have been described in IE Bulletins 79-02 (Reference 7), 79-06A (Reference 8), 79-07 (Reference 9), 79-13 (Reference 10) and 79-14 (Reference 11) and are being reviewed by IE and/or NRR. The status of these potential safety problems is also included in this evaluation.

Discussion

Technical Specifications

Salem Unit 1 completed its first cycle of operation on March 31, 1979 and immediately began preparing for initiation of Cycle 2 in June 1979. The only items identified by the licensee for review for Cycle 2 related to the reload core which was proposed to consist of 40 new Westinghouse 17 x 17 fuel assemblies. Two of these assemblies are of the optimized fuel assembly design as part of the Westinghouse "Optimized Fuel Assembly Demonstration Program." The licensee has reviewed the "as loaded" Cycle 2 core in relation to the Cycle 1 core that was reported in the FSAR and, as a consequence, propose the following changes to the Technical Specifications for Cycle 2 operation:

1. Increase in radial peaking factor (F_{xy}).

2. Revision of the normalized heat flux hot channel factor (K(z)) curve third line segment.

3. Restriction of axial flux difference for the first 2700 MWD/MTU.

4. Revision of the nuclear enthalpy hot channel factor ($F_{\Delta H}$) to take credit for currently approved rod bow penalty.

Our review of these proposed Technical Specification addresses the licensee's earlier request to change the $F^N_{\Delta H}$ limit to account for the reduction in departure from nucleate boiling ratio due to fuel rod boiling (Reference 12).

Grid Strap Damage

The reloading outage for Cycle 2 has been extended far beyond the original schedule because of several reasons. The first unexpected problem arose when the licensee observed that some of the Cycle 1 fuel assemblies were damaged when removed from the core. This problem and its satisfactory resolution is addressed in our overall evaluation of the Cycle 2 core.

Broken Rodlets

The outage and review of Cycle 2 operation have been extended further because of structural failures that were observed in six RCCA's that contained eight broken rodlets. The licensee discussed this problem in detail with the staff (Reference 13) and, subsequently, Westinghouse provided its findings and guidelines for early detection of dropped rodlets (Reference 14). Because of the uniqueness of this problem, we have reviewed the analyses performed by Westinghouse and the licensee's resolution for Cycle 2 to assure safe operation of the Cycle 2 core.

IE Bulletins

As part of its continuing review of nuclear plants that have been licensed to operate, the Office of Inspection and Enforcement identifies safety problems that may be of generic nature to all or specific types of nuclear reactors. Five of these Bulletins have been considered to be pertinent to the Salem Unit No. 1 plant and of sufficient potential to warrant discussion or evaluation before permitting initiation of Cycle 2.

IE Bulletin 79-02 - Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts

As the result of its concern over structural failure of piping supports of safety equipment, the NRC has described actions required of the licensee to determine if this potential exists at Salem Unit 1. Before achieving Mode 4, Hot Shutdown, the licensee has agreed to meet the requirements of IE Region I to:

1. Complete a test program to verify correct installation of wall and ceiling mounted concrete anchors in safety-related systems in inaccessible areas,

2. Submit a test program outline for testing of anchor bolts and base plates in accessible areas. (This testing s expected to be completed by November 15, 1979.), and

3. Perform an evaluation detailing the basis for resumption of operation (Reference 15).

The licensee responded to these requirements by letter of September 24, 1979 (Reference 16) and is presently performing the verification tests.

IE Bulletin 79-06A - Review of Operational Errors and System Misalignment Identified During the Three Mile Island Incident

This Bulletin identifies certain actions to be taken by the licensee to review design and operational aspects of Salem Unit No. 1 that may be similar to those that were in affect at TMI. The licensee responded to this Bulletin by letters of April 25, 1979 (Reference 17), May 11, 1979 (Reference 18), July 13, 1979 (Reference 19) and August 14, 1979 (Reference 20). These responses have been reviewed by a special NRC task force and have been found to be acceptable to permit return of Salem Unit No. 1 to power. This task force is continuing to review specific long-term provisions of this Bulletin.

IE Bulletin 79-07 - Seismic Stress Analysis of Safety Related Piping

In the course of evaluation of certain piping designs, significant discrepancies were observed between the original piping analysis computer code used to analyze earthquake loads and a currently acceptable computer code developed for this purpose. The licensee was notified on August 28, 1979 (Reference 21) that, prior to achieving Mode 4, hot shutdown during

Cycle 2, the licensee must provide the basis for return to service prior to completion of all the requirements of this Bulletin. The licensee responded by letters of September 21, 1979 (Reference 22) and of October , 1979 (Reference 23). Our review of these responses is included in this evaluation.

IE Bulletin 79-13 - Cracking in Feedwater System Piping

After discovery of cracking in the feedwater lines of other Westinghouse steam generators, the NRC required the licensee to determine if similar problems existed at Salem Unit No. 1. In response the licensee made the necessary inspections and informed the NRC that cracks had been found (Reference 24). Additional information was provided by letters of June 14, 1979 (Reference 25) and August 24, 1979 (Reference 26) and during a meeting with the staff on July 12, 1979. The licensee also provided the staff with samples of the faulty sections of the steam generator for analysis. Our safety evaluation of the licensee's actions is enclosed.

IE Bulletin 79-14 - Seismic Analysis for As-Built Safety Related Piping Systems

By means of this Bulletin, the NRC requested the licensee to take certain actions to verify that seismic analyses are applicable to plants as-built. The licensee is implementing the requirements of this Bulletin as part of its program to respond to Bulletin 79-07. By letters of August 16, 1979 (Reference 28) and September 14, 1979 (Reference 29) the licensee described the field walks of safety-related piping systems that are being made. Sample walks have confirmed that actual configurations conform to the stress isometric drawings. Inspection of all inaccessible areas will be performed before the plant returns to power.

Loss of Eddy Current Template Plug Assembly

Eddy current testing of the steam generator tubes was performed during the reload outage for Salem Unit No. 1. At the conclusion of this testing, one of 24 plug assemblies brought onsite for this purpose could not be accounted for and is assumed to be lost inside the primary coolant system. The licensee notified the staff to this effect on May 9, 1979 (Reference 27).

An analysis of this potential problem by Westinghouse indicates that the plug is of insignificant mass and physical size to cause any effect upon the plant safety analysis or operation (Reference). Tests have shown that the 21 gram plug (approximately 2 inches long) and 1 inch in diameter) will undergo mechanical disintegration from the turbulence of the Primary Coolant System and will undergo thermal decomposition at the operating temperature of the Primary Coolant System. A chemical analysis of the residual components of a plug heated to a Reactor Coolant System operating temperature would not be detrimental to the integrity of the system operations or equipment.

Our review of this problem indicates that if a plug is located in the hot leg area of the steam generator it will remain there until it has been mechanically fragmented or decomposed to such an extent that it will pass through the steam generator tubes. The size of the plug will probably be reduced by both mechanisms while the Primary Coolant System is being heated through operation of the coolant pumps during startup for Cycle 2. The composition of the residue from the plug has been shown by Westinghouse to be a low density pliable mass with a maximum weight of approximately 5 grams. The fate of such a semi-solid mass would depend on whether it would adhere to a portion of the Primary Coolant Boundary before it was converted into colloidal state as an emulsion. The total mass and size of the eventual residue is considered to be too small to result in a blockage in the Primary Coolant System such as has been analyzed in the FSAR. Therefore, we agree that the possible presence of a template plug in the steam generator does not pose a safety problem during Cycle 2.

Evaluation

1. Proposed New Technical Specification Changes

Nuclear Design

The Cycle 2 loading consists of 36 region 1 fuel assemblies (16.6 MWD/MTU average burnup), 60 region 2 fuel assemblies (17.0 MWD/MTU average burnup), 57 region 3 fuel assemblies (12.4 MWD/MTU average burnup), and 40 fresh region 4 fuel assemblies. The grid damage and rodlet drops discovered during the refueling outage prompted reanalysis of the core. The results of the as-loaded core analysis are discussed here.

Cycle 2 operation is designed with a peaking factor envelope limit of 2.32. The large break LOCA analysis provided in Reference 29 was performed with a peaking factor of 2.32 using the February 1978 model. This has been reviewed and approved by the staff (Reference 30).

For Cycle 2 operation, the licensee provided a reanalysis (Reference 2) of the small break LOCA to justify a revised third line segment of the normalized operating envelope. This was performed with approved methods and is acceptable.

An "18 cases" analysis was provided by the licensee to show that the worst peaking factors encountered for postulated load follow maneuvers during Cycle 2 are within the bounds of the proposed normalized operating envelope. The analysis was performed with a radial peaking factor (Fxy) of 1.65 and was done according to the methods described in Reference 17. The results are acceptable for axial offsets (I) of $\pm 5\%$ from the target band.

For Cycle 2, the licensee proposed a Technical Specification which limits the axial flux difference to less than positive 7.5 percent for the first 2700 MWD/MTU. The licensee asserts that analysis performed for the first 2700 MWD/MTU. The licensee asserts that analysis performed by approved methods shows that although the target flux difference at BOC is greater than 7.5 percent, it will be necessary to limit the axial flux difference to less than 7.5 percent to assure that the Fq limit of 2.32 is maintained. We find this acceptable for Cycle 2.

The rod bow penalty Technical Specification has been updated for Cycle 2 and conforms to the provisions of Reference 34. We find this acceptable.

For Cycle 2, the licensee is loading two Westinghouse 17 x 17 demonstration "optimized fuel assemblies." These assemblies are similar to current 17 x 17 design except that zircaloy grid straps are used and the fuel pins are slightly smaller in diameter. Loading criteria developed by Westinghouse, based on nuclear and thermal-hydraulic analyses and presented in Reference 16, require that the demonstration assemblies be placed in locations such that $F_{\Delta H}$ is at least six percent lower and Fq is at least 0.10 peaking factor units lower than the maximum allowed for standard assemblies. For Cycle 2 at Salem, the locations of the demonstration assemblies are at the core periphery where both these criteria are met. One assembly is instrumented with a thermocouple and the other with a movable incore flux detector. The licensee, in cooperation with Westinghouse, will follow the fuel surveillance program proposed in Reference 35. Because the use of the demonstration assemblies is limited to two assemblies loaded in low power regions of the core, we find their use acceptable for Cycle 2.

Fuel Design

With the exception of the two optimized fuel assemblies, the Cycle 2 reload fuel assemblies are of the same mechanical, nuclear and thermal hydraulic design as the Cycle 1 fuel assemblies. The Cycle 2 fuel rod internal pressure design criteria limit the internal pressure of the lead rod in the reactor to a value below that which could cause (1) the diametric gap to increase due to outward cladding creep during steady state operation and (2) extensive DNB propagation to occur. The NRC has accepted this design basis (in Reference 31). Calculations of the clad flattening time predict no clad flattening during Cycle 2.

2. Fuel Assembly Grid Anamolies

During the refueling operation at Salem, it was noted by the licensee that some of the assemblies that were removed had suffered grid mechanical damage. This was reported to the NRC in Reference 5. Subsequent to this discovery, all fuel assemblies were removed from the core for examination. The degree of the damage to the grid straps was classified in three categories: small pieces missing (15 assemblies), grid material ripped and laid over (5 assemblies), larger sections missing and fuel pins exposed (11 assemblies). No damage to the fuel pins was observed. A total of 31 assemblies suffered some grid damage.

The damage appears to be the result of corner to corner interaction of the grid straps of diagonally adjacent fuel assemblies during the vertical loading and unloading movements. No correlation of the damage to core location, grid elevation, or manufacturing and shipping batches has been identified.

The licensee and the fuel manufacturer established the following guidelines for Cycle 2 for reloading damaged assemblies: (1) those assemblies with full width pieces missing will not be reloaded for Cycle 2; (2) those assemblies with deformed edges and those with chips missing will be reloaded with special procedures to prevent further damage.

Westinghouse has noted similar damage of this type in several other plants, However, in none of these cases were the number of assemblies damaged as great as at Salem. Examination of some of the damaged assemblies indicated that they had operated through the previous cycle with no detrimental operational effects.

The nuclear and thermal-hydraulic effects of operation during Cycle 2 with 19 assemblies which have bent or chipped grid spacers is expected to be minimal. The loss of Inconel metal from the grids is insufficient to have a discernible effect on the core neutronics.

It should be noted that there was no observable grid compression or deformation. The damage was restricted to the grid straps and tabs. Nevertheless, if any minor deformations had been observed, they would have resulted in effects that would have been bounded by the rod bow considerations which were included in the Salem reload analysis and Technical Specifications and have been found to be acceptable.

There was some concern that pieces of grid strap which were not recovered would result in either flow blockage of an assembly or in jamming to prevent scram of an individual control assembly. With respect to flow blockage, the Salem FSAR (Reference 32) describes the results of analyses of complete blockage of an assembly nozzle and of partial flow blockage in the subchannels. For complete blockage of an assembly inlet nozzle, the analysis with the THINK-IV code shows that the flow is restored to normal within 30 inches of the nozzle. For those locations where the flow is disrupted, the DNBR does not approach 1.30 at full power conditions because these are not the peak power regions of the core. Examination of all damaged assemblies shows that a total of approximately 25 square inches of grid material was broken off. The licensee estimates that after recovery of some of the larger pieces, no more than seven pieces larger than two square inches each remain somewhere in the RCS. All these pieces together would be insufficient to totally block an assembly nozzle. Therefore, this event is considered impossible.

The FSAR also describes tests with partial flow blockage in the coolant channels, which show that with as much as 41 percent of the subchannels blocked, flow recovers to normal within five inches of the blockage. It is estimated that at Salem for full power steady state conditions, a reduction in local mass velocity of approximately 70 percent would be required to reduce the DNBR to 1.30. The mass velocity effect on the DNB correlation was based on the assumption of non-turbulent flow along the channel length. In reality, a local flow blockage is expected to promote turbulence which would lesson the effect on DNB. For pieces of grid strap which are free in the RCS and large enough to cause even minor blockage, the most likely place for the blockage would be the bottom nozzle or the first grid assembly elevation. These are relatively low power locations. Because of the limited effects of flow blockage from small pieces of grid strap in the fuel channels and because this blockage is expected at low power elevations, we believe the consequences of this type blockage do not endanger the public health and safety.

With respect to jamming of a control assembly to prevent scram, the likelihood of a piece lodging where it could cause a problem is extremely remote. Most of the chips would be expected to settle out in the stagnant regions of the lower plenum. However, if a piece were entrained in the RCS flow and lifted through the assembly inlet nozzle, it would be very unlikely to complete the torturous path up through the fuel assembly to the upper internals where it could lodge to interfere with a scram. Technical Specifications call for periodic exercising of control assemblies which would alert the operators of any control rod binding should it occur. In addition, all accident and transient analyses which result in reactor scram are done assuming that the most reactive control assembly does not scram. Because of the very small likelihood that an unrecovered piece could prevent scram, because rods must be periodically exercised, and because the consequences of one stuck control assembly are acceptable in all accident and transient analyses, we find that it is acceptable to operate for Cycle 2 without recovery of all small grid fragments.

3. Safety Evaluation of Broken Rodlets

During the current refueling at Salem, some reactor control cluster assemblies (RCCAs) were observed with individual rodlets which had broken from the main assembly. This was reported to the NRC in Reference 6. Subsequent to this initial discovery, all RCCAs in the Salem core were inspected. Six RCCAs with a total of eight detached rodlets (four RCCAs with one broken and two RCCAs with two broken) were found. The detached rodlets remained inserted in their respective fuel assembly guide tubes.

Examinations of the failed RCCAs have shown that the failures occurred in the threaded area of the (female fitting) fingers into which the (male fitting) rodlets are threaded, torqued and pinned. The failures were complete circumferential cracks in the fingers at a location adjacent to the topmost threads of the rodlet endpiece. All the dropped rodlets were traced to two receiving lots of fingers from a manufacturing subcontractor. All the fuel assemblies containing dropped rodlets and all the RCCAs (25) with fingers from the two suspect receiving lots have been removed from the reactor.

3.1 Materials Considerations

Each of the 53 RCCAs in a Westinghouse reactor with 17 x 17 fuel contains 24 individual rodlets for a total of 1272 rodlets. Earlier Westinghouse 15 x 15 fuel designs use RCCAs with 20 rodlets each. In both cases, the finger designs are similar with only slight variations in dimensions. The past performance of Westinghouse RCCAs has been satisfactory. A total of 1382 RCCAs are in 33 operating plants with individual service times that range from a few months to 140 months. Through December 1978, only ten RCCAs have experienced some operational problem. In six of these, the rodlets became detached from the spider hub due to vane separation. This separation has been attributed to faulty braze joints. Two RCCAs required repair due to galling of single rodlets in each assembly. One RCCA was discharged due to bent rodlets. One RCCA experienced a single rodlet separation in a manner somewhat similar to Salem Unit No. 1. About half of these reported events occurred during the initial cycle of reactor operation while the remainder occurred randomly over a period of 2 to 9 years. In general, each event involved only 1 or 2 RCCAs.

A review of the design of the finger at the location of the break showed that worst case loads were no more than 20 percent of the design values. In fact, the point of failure is designed as one of the strongest points in the RCCA. Thus, the failures are not likely due to stress or fatigue.

All failures occurred in fingers in the outer row. Within the outer row of fingers, the failures occurred in random positions. No failures occurred in fingers of the inner rows. The physical positions of the RCCAs with failed fingers were random with respect to core location.

In an effort to determine the cause of the failures, Westinghouse reviewed manufacturing records of the affected fingers. This review included materials records, procurement records, deviation records and manufacturing processes. The data is presented in the attachment to Reference 8. The results of this review of materials records and procurement records determined the following:

1) All failed fingers were machined from two material heat lots and were contained in two sequential receiving lots of outer fingers from one supplier.

2) Many other receiving lots of both outer and inner fingers which did not exhibit failures were also machined from the same heat lots by the same supplier.

3) The total population of outer fingers in Salem Unit No. 1 for Cycle 1 was machined from three material heat lots and was comprised of 11 receiving lots from three different suppliers.

The conclusions drawn from the materials records are that the problem is limited to the two receiving lots of fingers at Salem and does not extend generically to other Westinghouse plants.

Westinghouse asserted that review of the deviation records associated with finger machining and spider assembly showed nothing of significance. The review of the manufacturing processes for finger machining and spider assembly was not conclusive but provided these observations:

1) Any contaminants left by the finger machining process would probably be cleaned out by the high temperature and vacuum applied to the finger during brazing to the spider.

2) During final assembly of rodlets to fingers, retapping of threads and shoulders is sometimes required for final fitting and a contaminant could have been introduced at this time. The cleaning process is not repeated after retapping. Since the time of manufacture of the Salem fingers in 1975, manufacturing change notices were issued by Westinghouse to change tapping specifications so that fewer fingers needed retapping and also to remove a certain threading lubricant from the assembly area.

The conclusion from the review of the manufacturing processes are that an undetermined contaminant may have been introduced in the threaded area of the finger after initial cleaning along with high residual stresses but that changes in the fabrication process have since eliminated the problem.

It should be noted that one difference between the fabrication process for 15 x 15 RCCAs and 17 x 17 RCCAs was found. It is sometimes necessary to retap the fingers to allow proper fit of the threaded rodlet into the finger. Because of a slightly shallower threaded area on 17 x 17 fingers, a bottoming tap is used to retap the threaded area rather than a tapered tap. This could result in increased stress concentration in the 17 x 17 fingers. However, due to the large design margin to stress at this point, and the absence of failures in fingers from other receiving lots, it was concluded that the effects of this difference in tapping would be insuffucient to cause the failure.

As part of the materials investigation, hot cell work was performed on two RCCAs removed from Salem Unit No. 1. After Cycle 1, RCCA R-31, which contained two failed fingers, and RCCA R-37, which contained no failed fingers and no fingers from the two suspect receiving lots, were examined. Three damage types were discovered and the damage was all characterized as stress-corrosion cracking. The three finger damage types were described as: (1) larger circumferential cracks in the top threaded region; (2) minor axial cracks in the thread area; and (3) local cracking in the shoulder area.

Eleven of sixteen outer fingers from R-31 were sectioned. Only three were clear of damage. Of the remaining eight that were sectioned, three (in addition to the two that had failed during cycle 1, one additional rodlet finger failed out of the reactor prior to hot cell testing) had failed completely, six exhibited 20 to 80 percent circumferential cracking in the top threaded area, and three exhibited local cracking in the shoulder area. All four middle fingers were clear. Of four inner fingers, two were clear and two exhibited local cracking in the shoulder area.

RCCA R-32, which contained no fingers from the suspect lots, showed nine of nine sectioned outer fingers clear of damage, four of four middle fingers clear, and three of four inner fingers clear. One inner finger showed local cracking in the shoulder area.

According to Westinghouse, tests for elevated chloride levels in the outer finger of R-32 appeared to show greater concentration than the inner fingers; however, the results of this testing were not conclusive.

Further metallurgical examination of the crack surfaces of the circumferential cracks indicated that the cracks were "old" and had probably occurred early in the cycle. Further evidence of this is provided by the appearance of the flux tilt early during Cycle 1 operation, and with hindsight, by examinations of Cycle 1 flux maps. As an estimate of the time frame for failure due to stress corrosion, Westinghouse estimated that with a saturation concentration of chloride ions at 550°F, with stress concentrations in the threaded area, failure could occur in less than one hour.

The conclusions reached by Westinghouse and the licensee with respect to the failed RCCA fingers are:

1) Failures do not represent a structural inadequacy or generic design weakness.

2) Failures are the result of stress corrosion cracking and were contained within the two receiving lots of outer fingers.

3) Indications of stress corrosion cracking on other than the two receiving lots are located in the shoulder area, are of a different composition and severity, and would not lead to dropped rodlets.

4) A review of the flux maps of operating reactors and successful refueling of two 17 x 17 cores shows that no positive evidence of broken rodlets exists for other plants.

5) A review of Salem Cycle 1 flux maps shows that dropped rodlets occurred prior to low power operation and were present throughout Cycle 1.

6) Elimination of all RCCAs containing fingers from the suspect lots should prevent recurrence.

The staff agrees that the evidence presented by the licensee supports the conclusions stated above.

3.2 Nuclear and Thermal-Hydraulic Considerations

Calculations were performed by the licensee and by Westinghouse to provide estimates of the nuclear and thermal-hydraulic effects of broken rodlets. The information was provided in Reference 8 and in various telephone conversations with the licensee and with Westinghouse.

The reactivity of a dropped rodlet in the core is estimated to be worth about 10 pcm per dropped rodlet. Using this estimate, and the value for the excess shutdown margin (SDM) available during cycle two of 0.5% k/k, at least 50 rodlets dropped randomly into the core would be required to cancel the excess SDM. The value of excess SDM is calculated assuming the rapid cooldown of the moderator due to a steamline break and failure of the most reactive RCCA to scram.

Relative to shutdown margin requirements to accommodate the postulated steam line break at end of cycle, the licensee has demonstrated ample shutdown margin to accommodate all other postulated transients. The combination of low probability events required to potentially endanger the public health and safety are: (1) large steamline break; (2) most reactive RCCA stuck; and (3) more than 50 rodlets dropped. As discussed later in this evaluation, core surveillance would make it unlikely for such a large number of dropped rodlets to go undetected. Because of the unlikelihood of the combination of low probability events and the likelihood of detecting 50 dropped rodlets, we believe that loss of shutdown margin due to dropped rodlets is not a significant safety concern.

Cycle 1 was operated with excess SMD of 1.60% k/k. This was equivalent to the worth of at least 160 rodlets. Thus with respect to SDM, safe operation during Cycle 1 was not jeopardized with eight dropped rodlets.

The presence of a detached rodlet in the core could be of concern with respect to mechanical movement of a loose part. In the case of Salem, all of the detached rodlets remained in the guide tubes of the respective RCCAs. It is expected that because of the upper guide structure templates, a rodlet which fell from a withdrawn position would be guided without binding into its respective RCCA

RCCA guide tube. The hydraulic uplift force on an individual rodlet due to normal reactor coolant flow through the guide tubes is estimated by Westinghouse to be less than one half the force required to lift a rodlet from its fallen position. Thus, the staff concludes that there is no danger of loose rodlets moving about in the reactor coolant system.

The effectiveness of the Technical Specifications for detection of power anomalies such as dropped rodlets was demonstrated during Salem Cycle 1 operation when a flux tilt in excess of the Technical Specification allowances was encountered. Although the cause of the tilt was not discovered at that time, in retrospect the cause has been identified as dropped rodlets. Flux maps taken at that time also showed the presence of the dropped rodlets. In any case, the licensee was required to analyze the tilt to show that safety limits were not jeopardized. Flux maps taken at that time also show the presence of the dropped rodlets.

Determination of values of Fxy, Fq and $F_{\Delta H}$ are required to be made at least once every 31 days of operation. The values of these peaking fators are determined from incore instrumentation measurements which would include the effects of the flux depressions due to the dropped rodlets. If the allowed values of these peaking factors are exceeded during power operation, power must be reduced. The accident and transient analyses are valid only if the peaking factor limits are maintained. Since the peaking factors can be measured regardless of the presence of dropped rodlets, it is possible to maintain the core in a safe condition by observing the current applicable Technical Specifications.

3.3 Augmented Surveillance and Startup Program

All evidence from Cycle 1 at Salem indicates that the failure of the fingers resulting in the dropped rodlets occurred prior to going to power. With hindsight, hot zero power flux maps at beginning of Cycle 1 show dropped rodlets. Also, Westinghouse estimated that with saturation chloride levels at 550°F, the failures could occur within less than one hour.

During startup testing, the licensee is required to measure shutdown margin and to measure the critical soluble boron concentration for comparison with calculated values. Ideally, it should be possible to detect dropped rodlets at this time. The accuracy of the titration methods used for measurements of soluble boron concentration is estimated to be approximately ±10 ppm boron. This converts to approximately ±0.1 k/k reactivity. This measurement error is about 20% of the magnitude of the reactivity worth of 50 dropped rodlets. The licensee is required by Technical Specifications to report deviations of greater than ±100 ppm of the design value of critical boron concentration. This would detect the presence of approximately 100 randomly dropped rodlets. And although not required by Technical Specifications, a discrepancy of ±50 ppm from design is used by the licensee as the criteria for initiating a design review. This criteria would detect the presence of approximately 50 dropped rodlets.

The licensee has also submitted (Reference 33) an augmented surveillance program for Cycle 2 startup to detect dropped rodlets. Flux map analysis will pay particular attention to flux depressions. Acceptance criteria for flux maps require discrepancies of less than $\pm10\%$ of design for assembly powers greater than 0.9 nominal and $\pm15\%$ of design for assembly powers less than 0.9 nominal. If the design acceptance criteria for flux maps is exceeded, the licensee will measure rod worths to the N-1 condition. In this way, if flux maps indicate a problem, measurements on individual RCCAs would be used to localize possible dropped rodlets.

In addition to the flux maps, the licensee will continuously monitor certain plant parameters to detect any changes that might indicate a dropped rodlet: 1) the core reactivity computer has sufficient sensitivity to detect a change of approximately 10 pcm (the worth of one rodlet); 2) the primary coolant temperature instrumentation is expected to detect the occurrence from one to three dropped rodlets due to moderator temperature feedback; 3) control rod motion, turbine load, xenon, boration, and dilution will be monitored to separate intentional reactivity changes from unexpected deviations. Any indications of dropped rodlets will be reported to the NRC. The staff agrees that the additional startup tests and augmented surveillance program proposed by the licensee provide sufficient assurance that failure of a large number of rodlets will be detected.

These additional tests and the augmented startup program are in addition to the standard physics startup test program which has also been reviewed. Low power physics include: boron endpoint, isothermal temperature coefficient, rod worth and flux map measurements.

The high power physics test include power coefficient and flux map measurements. Acceptance and design criteria as well as remedial actions for these tests have been approved. The staff considers this total program to be appropriate and adequate.

4. Seismic Stress Analysis of Safety-Related Piping (IE Bulletin 79-07)

PSE&G has used PIPDYN II computer code for pipestress analyses. The computer analysis involved calculation of piping responses due to X-component earthquake, Y-component earthquake, Z-component earthquake, X and Y earthquake, and Y and Z earthquake. During the X and Y and Y and Z earthquake evaluation, however, the intramodal piping responses were inadvertently calculated by use of the algebraic summation method. This is considered unacceptable as it may predict unconservative results in the seismic piping analysis. This code with the intramodal summation method was used in the seismic analyses of most of the safety-related systems at the facility. The licensee has identified the seismically analyzed (Seismic Category I) systems at the facility analyzed with PIPDYN II and the algebraic summation technique. It has also identified portions of the Control Air System as the only system not seismically analyzed (i.e., static method). Furthermore, the licensee has reported the results of reanalyses using an acceptable earthquake response summation technique. This latter technique consists of utilization of the individual X, Y, and Z earthquake responses which were previously computer calculated using PIPDYN II and the hand calculation of the root-sum-square (SRSS) of intramodal responses due to the three components of earthquake loading.

We have evaluated the results of all the methods of pipe stress analysis previously utilized and used in the reanalyses for the facility. Technical information required for this evaluation is provided in the licensee's submittals of August 28, 1979, September 21, 1979 and October 11, 1979.

1) Systems

The following 15 systems were identified by the licensee as having been analyzed with PIPDYN II with the algebraic summation technique:

Residual Heat Removal
Reactor Coolant
Safety Injection
Steam Generator Feedwater
Component Cooling
Service Water
Auxiliary Feedwater
Diesel Generator Starting
Air and Fuel Systems
Chilled Water
Chemical and Volume Control
Control Air System
Steam Generator Blowdown Systems
Spent Fuel Cooling
Main Steam
Containment Spray System

The licensee has reanalyzed all 823 pipe stress problems originally involved in the algebraic summation calculation. In addition, the licensee has stated that the pipe stress problems, which were analyzed by hand calculation, did not sum earthquake responses algebraically and are acceptable.

The licensee's letters of October 11, and October 18, 1979 entails three phases of program as described below:

Phase I

Prior to entering Modes 3 and 4, the following work will be accomplished:

(a) Completion of pipe stress analysis (for both OBE and DBE) on safety-related systems required for safe shutdown.

(b) Re-evaluation of the associated supports, nozzles, and penetrations, within the inaccessible area.

(c) Re-evaluation of the supports, nozzles, and penetrations for entire Auxiliary Feedwater System.

(d) Re-evaluation of the supports for the Reactor Coolant System Pressure Boundry.

(e) Field modification to supports and penetrations evaluated in (2), (3) and (4) that fail to meet the criteria stated in September 21, 1979 submittal. Field modification to nozzles which fail to meet manufacturer's acceptance criteria.

(f) Re-evaluation of the supports, nozzles, and penetrations of the following systems:

    (1) High pressure safety injection using the Chemical and Volume Control System.

    (2) Low pressure safety injection using the Safety Injection System.

    (3) Main Steam System up to the isolation valves to include the steam supply to the steam driven auxiliary feed pump.

    (4) Containment Spray and Recirculation.

(g) Field modification to supports and penetrations evaluated in (f) that fail to have a factor of safety of at least 2. Field modification to nozzles which fail to meet manufacturer's acceptance criteria.

Phase II

Prior to entering Modes 1 and 2 the following work will be accomplished:

Field modification and corresponding modifications associated with IE Bulletin 79-02 to supports and penetrations evaluated in item f of Phase I that fail to meet the design criteria as stated in September 21, 1979 submittal. Modifications will be made within the time constraints of the action statements of the Technical Specifications if re-evaluation shows that system operability is affected.

Phase III

The licensee shall complete reanalysis of the remaining pipe supports, nozzles and penetrations outside containment and shall propose a schedule for implementation of all identified modifications, both within 60 days of the date of plant startup.

For each modification identified as a result of reanalysis of the supports outside containment after resumption of facility operation, when the overall margin of safety of the support to ultimate capacity is determined to be less than 2, the NRC shall be notified within 24 hours after making each such determination. The affected system shall be considered inoperable as that term is used in the facility Technical Specifications until the necessary modifications are implemented within the time frame allowed by the facility Technical Specifications.

Of the 823 pipe stress problems re-evaluated, requirement for hardware modifications have been identified on 248 individual supports to bring the pipe stresses in the inaccessible area and auxiliary feedwater system within allowables. It has been revealed, during reanalysis that most of these modifications can be attributed simply to the original support design errors, rather than due to use of the algebraic summation method. Classification of these unacceptable supports together with the proposed modifications are typically as follows:

(a) 97 U-bolts used in most systems as anchors or guides were inadequate to withstand lateral loads and moments. Add structural steel or use heavier material.

(b) 34 structural steel members used in most systems contain certain members that could not withstand torsional or bending moments. Add steel members.

(c) 57 straps in most systems used as anchors or guides were inadequate to withstand lateral loads and moments. Add structural steel, plates and heavier material.

(d) 31 welds in supports in most systems were overstressed. Strengthen the weld itself and/or add bracing to relieve the stresses on the weld, or eliminate that particular support through stress calculation.

(e) 22 trunions used mostly to anchor 6" diameter piping in component cooling system were overstressed. Add structural steel beams, straps, gusset plates and/or additional welds.

(f) 4 undersized rods. Replace with correct size rods.

(g) 1 undersized snubber in residual heat removal system. Replace with one of proper size.

(h) 2 improperly embedded anchor bolts. Replace base plate and use a new bolt pattern with additional bolts to carry the design load. Also add new bracing.

2) Verification of Analysis Methods

We have reviewed the acceptability of the analytical methods which are currently a basis for the facility piping desing. The licensee has identified the following computer code/analysis methods as applicable:

PIPDYN II (used only for response calculations for individual earthquake components)
Static Analysis Methods

PIPDYN II

In response to IE Bulletin 79-07, the licensee has submitted documentation of the positions of the computer code which was used in the piping reanalysis of Salem 1. The code is called PIPDYN II, which originated at the Franklyn Institute Research Laboratory, (FIRL).

FIRL has stated that this code performs response spectrum and time-history analysis and that it calculates intramodal and intermodal responses according to the provisions of Regulatory Guide 1.92. A review of the code listing, as well as direct communications with FIRL personnel, has confirmed this.

The FIRL is presently solving a set of NRC designed benchmark piping problems, using the response spectrum analysis method. A preliminary comparison of FIRL and NRC solutions indicate good agreement.

The licensee has also submitted recently a piping problem and solutions for confirmatory calculations by the Brookhaven National Laboratory. These calculations are scheduled to be completed in the immediate future.

Based on these considerations we find the use of the present version of the code PIPDYN II provisionally acceptable for seismic analysis by the response spectrum method.

Static Analysis

Some of the 1-1/2" or smaller safety-related low temperature field run piping in the control air system at Salem Unit 1 was analyzed using simplified static methods. For such piping the seismic support spacing was determined by assigning a rigid frequency (say 30 cps) to an equivalent simply supported straight beam. The approach for support sizing was to support the system rigidly at specific intervals already determined. The support loads and piping stresses were then calculated by the static load method assuming the highest peak acceleration over the entire frequency ranges of the floor response spectra.

3) Reanalysis Methods and Results

The safety-related piping systems at Salem Unit 1 have been reviewed to determine the method of analyses. Eight Hundred and Twenty-three (823) computer stress problems of safety-related piping have been identified where the analysis used the computer code PIPDYN II which used an algebraic intramodal summation of responses to earthquake loadings. The problems where an algebraic intramodal response combination technique was used in the design have been reevaluated using an acceptable method. The method uses the individual X, Y and Z earthquake responses previously calculated by PIPDYN II and then uses hand calculations to combine the above intramodal responses by the SRSS method.

The floor response spectra used in the reanalysis was the original amplified response spectra specified in the FSAR. The peaks in the amplified floor response spectra were broadened by $\pm 10\%$ to account for variation in material properties and approximations in modeling.

The piping systems were modeled as three dimensional lumped mass systems which included consideration of eccentric masses at valves and appropriate flexibility and stress intensification factors. The dynamic analysis procedures meet the criteria specified in the plant FSAR and are acceptable. The resultant stresses and loads from the reanalysis were used to evaluate piping, supports, nozzles, and penetrations.

All of the 823 PIPDYN II pipe stress problems have been reanalyzed and verified by the licensee's Quality Assurance Program. This reanalysis completed the entire scope of piping stress reanalysis. Based on the information provided for review, we find acceptable the procedures and methods used in reanalyzing these problems.

The reanalysis included those pipe stress problems involving the reactor coolant pressure boundary and the supports associated with those problems. Since the reactor coolant pressure boundary is inside containment and all of the supports which must be modified will be modified prior to startup, there is no potential for a loss-of-coolant accident in the event of a DBE.

The licensee has stated that I&E Bulletin 79-04, "Velan Valve Weights," presents no problem to the reanalysis program.

The pipeline support designs for affected system piping was inspected by the licensee to verify the location, orientation, support clearances, and support type. Any deviations that were identified are incorporated into piping reanalyses. These piping systems were also verified by the NRC Office of Inspection and Enforcement.

The pipe supports were reevaluated in cases where the original support design loading was exceeded as a result of piping reanalysis. In such cases, the support reevaluation has included the consideration of base plate flexibility and a verification of actual field construction of the support. Where concrete expansion anchor bolts were used, their capacities, without compromising the originally committed safety margin, were also included in the reevaluation.

There are approximately 5100 supports in the 15 safety-related systems involved in the reanalysis; of these, 3600 supports have been reevaluated. Among the supports reevaluated, 1548 supports are in the inaccessible area and 198 of them were identified to need modification based on the criteria stated in the September 21, 1979 submittal. The licensee has committed to complete all these modification prior to startup.

There are 297 supports associated with auxiliary feedwater system; of these a total of 50 supports have been identified to need modification based on the criteria stated in the September 21, 1979 submittal. The licensee has also committed to complete all these modifications prior to startup. In addition, there are also 771 supports outside containment which are associated with high and low head safety injection, containment and reciruclation spray, and main steam systems up to the isolation valves. All these supports have been reevaluated. Modifications to 147 supports which have been identified as failing to meet the design criteria as stated in September 21, 1979 submittal will be completed prior to entering Modes 1 and 2.

Based on the results to date, we expect other supports outside containment may be found that will not have a minimum factor of safety of 2 to ultimate, which is used as a criteria for support operability. However, if support reanalysis indicates this we will require the licensee to inform the NRC of the results of reanalysis within 24 hours and that the affected system be considered inoperable as specified in the facility Technical Specifications until the necessary modifications are implemented or a reanalysis assuming support failure is completed.

The licensee has examined nozzle loadings on 2 auxiliary feedwater pumps and 1 containment spray pump. The new forces and moments obtained from the reanalysis were included in the reevaluation. With regard to penetration loads, conservative hand calculation methods were used for simple configurations. For larger penetrations a finite difference computer program based on linear thin shell theory is used. Current results indicate no overstressed conditions. Effort is being continued on the nozzles and penetrations that are included in the reanalysis.

The licensee has committed to reevaluate all the nozzles and penetrations within the inaccessible area and for the entire auxiliary feedwater system prior to entering Modes 3 and 4. Field modifications will also be completed prior to entering Modes 3 and 4 for penetrations that fail to meet the criteria stated in the September 21, 1979 submittal and for nozzles which fail to meet manufacturer's acceptance criteria.

The licensee has also committed to complete reevaluation, prior to entering Modes 3 and 4, of penetrations and nozzles associated with high and low pressure safety injection, containment and recirculation spray, and main steam system up to the isolation valves. Field modifications will also be completed during the same period of time for penetrations that fail to have a safety factor of 2 to ultimate and for nozzles which fail to meet manufacturer's acceptance criteria. Furthermore, penetrations that fail to meet the criteria as stated in the September 21, 1979 submittal will then be modified prior to entering Modes 1 and 2.

Within 60 days of the date of plant startup reevaluations and field modifications, as appropriate, of the remaining penetrations and nozzles will be completed and the same operability requirement which is applied to supports will also be applicable to penetrations and nozzles.

The design and analysis of the supports and attached equipment are in accordance with the criteria specified in the plant FSAR.

The pipe break criteria of the FSAR was reviewed in connection with the possible effect of changes of the high stress point resulting from the reanalyses. Reanalysis completed thus far has not shown any requirement for postulations of additional break locations per the FSAR criteria. In cases where the FSAR criteria should be exceeded, new break locations will be postulated and protection provided as required. For inside containment, the same pipe break criteria for outside containment can be applied.

The piping systems and supports were designed to the allowable limits on ANSI B31.1. Components used in pipe supports (rods, U-bolts, clamps and bolts) were designed in accordance with ANSI B31.1 and MSS SP 58. Bolting of structural components associated with pipe supports were designed in accordance with AISC. Welded connections were designed in accordance with B31.1. The maximum loading conditions do not allow stress levels to exceed 24.0 KSI for fillet welds.

The safety-related piping systems supports and attached equipment, where the original analysis used an algebraic intramodal summation technique, have been, or are to be reanalyzed with acceptable methods. The procedures used in the support reanalyses and their results have been reviewed against the criteria in the FSAR and found acceptable.

4) Conclusions

The licensee has demonstrated that PIPDYN II is the only method of analysis used for the facility's safety-related systems which combines seismic loads algebraically. Safety-related piping systems analyzed with PIPDYN II have been reanalyzed with an acceptable method. Results of the reanalysis indicated that the pipe stress and equipment loads, after necessary modifications, will be acceptable when compared with the FSAR allowables and the manufacturer's specified load criteria.

The reevaluation of pipe stress problems indicated that modifications in 248 supports in the inaccessible area and auxiliary feedwater system were found to be necessary in order to bring the pipestresses to within allowable. These modifications are identified in Section 1, and the licensee will complete them prior to plant startup. Reevaluation will also be completed prior to startup for supports associated with high and low pressure safety injection, containment and recirculation spray, and main steam system up to the isolation valves. Any modifications required for these systems will then be completed either prior to startup, if a safety factor of at least 2 to ultimate does not exist, or prior to entering Modes 1 and 2, if the criteria stated in the September 21, 1979 submittal is not met.

Evaluation of the supports and schedule for completion of necessary modifications in the balance of the plant will be completed within 60 days of the startup. Further, in those cases where reanalysis exceeds code allowable, the staff requires that the criteria used to determine whether a factor of safety of 2 to ultimate does exist by linear elastic analysis techniques or no more than twice the rated load for snubbers. Use of Welding Research Council Bulletin #107 for evaluation of local stresses due to integral attachment is acceptable. Supports in accessable areas which exceed the factor of safety of 2 to ultimate will be considered as inoperable as defined in the Technical Specifications.

We reviewed the analysis techniques which are currently the bases for the facility's piping design. We have determined that the application of these techniques at Salem Unit 1 assures that safety-related systems will withstand the design basis earthquake. Although the reanalysis of supports outside containment is not completed, there is reasonable assurance that the facility can operate during the interim period until the reanalysis and any required modifications are completed without endangering the health and safety of the public. This assurance is based on the following factors:

(1) All safety system piping outside containment which was originally seismically analyzed with the PIPDYN II program has been reevaluated and, subject to modification, is acceptable.

(2) All piping and supports of the affected safety systems inside containment have been reevaluated and were found either acceptable as presently designed or will be modified as identified in this SER prior to startup. All the nozzles and penetrations in the same systems will be completely reevaluated and modified, if necessary, in accordance with the licensee's commitment as stated on its October 11, 1979 letter.

(3) Confirmation of input data through "as-built" verification provides assurance that analytical results are correct and significant "as-built" deficiencies repaired.

(4) The licensee has completed reevaluations and will implement necessary modifications prior to entering Modes 3 and 4 for the supports associated with auxiliary feedwater systems. The remaining reevaluations will also be completed prior to entering Modes 3 and 4 for the supports associated with high and low head safety injection, containment, and recirculation spray, and main steam system up to the isolation valves. Any necessary modifications for supports in these systems will be completed prior to entering either Modes 3 or 4 or Modes 1 and 2 in accordance with the licensee's letter of October 11, 1979. These systems and auxiliary feedwater system assure that ECCS systems and systems necessary for maintaining hot standby will be capable of withstanding a design basis earthquake.

(5) The licensee has committed to complete all the support reevaluation in accessable areas outside containment within 60 days of the date of plant startup.

(6) The probability of an earthquake exceeding the design basis earthquake during the 60 day period when the remaining support reevaluation is being completed is small and the licensee has committed to shut down the facility in the event of an earthquake which exceeds 0.01 g acceleration and inspect all piping, penetrations, supports and nozzles which have not been reanalyzed for both OBE and DBE.

(7) The NRC will require prompt notification of inoperable supports within 24 hours and either resolution by reanalysis of the piping system assuming a failed support or modification of the affected support, if reanalysis of a support indicates that a factor of safety of two to ultimate capacity does not exist (or snubber loading greater than twice rated capacity).

Based on the above, we conclude that the licensee has demonstrated why Salem Unit 1 can be operated for 60 days pending completion of reanalyses required by IE Bulletin 79-07.

5. Actions Taken to Eliminate Feedwater Piping Cracks

On June 19, 1979 in response to IE Bulletin 79-13, the licensee performed radiographic inspections of the steam generator feedwater nozzle/piping fitting welds and the adjacent area on all four steam generators. Crack-like indications were revealed by RT and UT in Nos. 11, 13 and 14 steam generator feedwater nozzle to pipe fitting weld areas. UT indications were noted in the same region of steam generator 12 but not shown by RT.

A meeting was held with the licensee on July 12, 1979 to discuss the following items regarding feedwater piping cracks:

1. Nature and extent of the cracking
2. Metallurgical evaluation of the cracking including identification of the mode failure
3. Stress analyses
4. Operating history
5. Feedwater chemistry
6. Corrective actions
7. Safety implications

The licensee's interim report on the feedwater line cracking is enclosed. In accordance with the requirements of IE Bulletin 79-13, the licensee performed volumetric examinations of all the feedwater piping welds with the exception of one inside of containment. In addition, magnetic particle examinations were performed of the auxiliary feedwater to main feedwater piping connections. No reportable indications were revealed from the results of the inspections.

The results of metallurgical evaluations by the licensee and their contractor of samples from loops 1, 2, 3 and 4 revealed cracks of a maximum depth of 0.120 inches in the region of the counterbase in the fitting in loops 1 and 4. Shallow cracks (0.025 inches) similar in nature to those in loops 1 and 4 were found in loops 2 and 3. The cracks were generally straight and not branched. Fractographic examination at how manification revealed beach marks. TEM examination, with great difficulty, indicated fatigue situations on the order of 1 to 3 micro-inches. The probable mode of failure was identified as corrosion assisted fatigue.

The licensee performed stress analyses in an effort to identify an anomalies which could cause the observed cracks. The analyses were:

1. Structural analyses using a 3D finite element model of the feedwater line including the effects of thermal, deadweight, and pressure (does not include stratification conditions). The licensee reports that results show the stresses are within the allowable code limits.

2. 2D finite element fatigue analysis of the feedwater nozzle/elbow configurations. The licensee reports that the results show an acceptable usage factor using the allowable cycles for a peak stress range from the ASME design S/N curves.

3. Frequency analyses of the feedwater line and steam generator. The licensee reports that the results of the analysis indicate that feedwater line/steam generator resonances is possible but consider this unlikely based on testing performed at the similar facilities that have been instrumented.

The piping fitting were removed and replaced on all steam generators. Any cracks identified by the liquid penetrant examination of the nozzle base ID or OD were removed and, if required, repaired. Repairs to fabrication related discontinuities in welds in the feedwater lines have been completed. The nozzle to fitting welds were fully radiographed and ultrasonically inspected following completion of the welding and stress relieving operations. The licensee has committed to perform radiography and ultrasonic examination of the nozzle to fitting welds at the next refueling outage. In addition the licensee has installed instrumentation to measure pressure, thermal and mechanical transients during startup operations.

We conclude that the actions taken and proposed augmented inspections and sufficient to insure that the piping integrity will be maintained. If the causes of cracking cannot be determined by the next refueling outage, we will then decide what further actions, if any, are necessary.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insigificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

Technical Specification Review

Since the Technical Specification changes presented by the licensee for Cycle 2 operation were developed as a result of analyses using approved methods and because none of the changes result in a significant decrease in the margin of safety, the staff finds the changes to be acceptable.

Potential Safety Problems Addressed by IE Bulletins

Responses to IE Bulletins 79-02 and 79-14 provided by the licensee have been determined by the Office of Inspection and Enforcement to be acceptable. Likewise, the responses to IE Bulletins 79-07 and 79-13 have been determined to be acceptably sufficient to permit restart of Salem Unit 2. Although initiation of Cycle 2 is not conditioned by the requirements of IE Bulletin 79-06A, the licensee has responded in an acceptable manner to all short term requirements of this Bulletin.

We have determined that the possible presence of an eddy-current template plug in the reactor coolant system does not pose a safety problem during Cycle 2.

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: October 30, 1979

1. Public Service Electric and Gas Company (PSEG) letter (Librizzi) to NRC (Schwencer) dated March 2, 1979.

2. PSEG letter (Librizzi) to NRC (Schwencer) dated April 30, 1979.

3. PSEG letter (Librizzi) to NRC (Schwencer) dated August 8, 1979.

4. PSEG letter (Schneider to NRC (Schwencer) dated August 9, 1979.

5. PSEG letter (Librizzi) to NRC (Grier) dated July 30, 1979 LER 79-44/03L-1.

6. PSEG letter (Librizzi) to NRC (Grier) dated July 30, 1979 LER 79-47/03L-1.

7. IE Bulletin 79-02 "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts," dated March 5, 1979.

8. IE Bulletin 79-06A "Review of Operational Errors and System Misalignments Identified During the TMI Incident," dated April 18, 1979.

9. IE Bulletin 79-07 "Seismic Stress Analysis of Safety Related Piping," dated April 14, 1979.

10. IE Bulletin 79-13 "Cracking in Feedwater System Piping," dated June 25, 1979.

11. IE Bulletin 79-14 "Seismic Analyses for As-Built Safety-Related Systems," dated July 2, 1979.

12. PSEG Letter (Librizzi) to NRC (Lear) dated October 7, 1977.

13. NRC July 17, 1979 Meeting Summary (W. Ross) Docket No. 50-272 dated July 25, 1979.

14. Westinghouse letter (Anderson) to NRC (Check) dated September 4, 1979.

15. NRC letter (Grier) to PSEG (Schneider) dated August 28, 1979.

16. PSEG letter (Schneider) to NRC (Grier) dated September 24, 1979.

17. PSEG letter (Schneider) to NRC (Grier) dated April 25, 1979.

18. PSEG letter (Schneider) to NRC (Grier) dated May 11, 1979.

19. PSEG letter (Schneider) to NRC (Grier) dated July 13, 1979.

20. PSEG letter (Schneider) to NRC (Grier) dated August 14, 1979.

21. NRC letter (Grier) to PSEG (Schneider and Martin) dated August 28, 1979.

22. PSEG letter (Schneider) to NRC (Grier) dated September 21, 1979.

23. PSEG letter (Librizzi) to NRC (Schwencer) dated October 11, 1979.

24. PSEG letter (Librizzi) to NRC (Grier) dated June 28, 1979.

25. PSEG letter (Schneider) to NRC (Stello) dated June 14, 1979.

26. PSEG letter (Schneider) to NRC (Grier) dated September 24, 1979.

27. PSEG letter (Librizzi) to NRC (Grier) dated May 9, 1979.

28. PSEG letter (Librizzi) to NRC (Grier) dated October 18, 1979.

29. PSEG letter (Librizzi) to NRC (Schwencer) dated February 15, 1979.

30. NRC letter (Schwencer) to PSEG (Librizzi) dated June 6, 1979.

31. NRC memo from D. Ross to D. Vassallo dated December 8, 1977.

32. Salem FSAR

33. PSEG letter (Librizzi) to NRC (Schwencer) dated August 3, 1979.

34. NRC memo from R. Tedesco to D. Vassallo dated March 28, 1979.

35. WCAP-9286 "Optimized Fuel Assembly Demonstration Program" dated July 1978.

36. WCAP-8385 "Power Distribution Control and Load Following Procedures" dated September 1974.

37. PSEG letter (Librizzi) to NRC (Schwencer) dated October 18, 1979.

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-272
PUBLIC SERVICE ELECTRIC AND GAS COMPANY,
PHILADELPHIA ELECTRIC COMPANY,
DELMARVA POWER AND LIGHT COMPANY, AND
ATLANTIC CITY ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 20 to Facility Operating License No. DPR-70, issued to Public Service Electric and Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees), which revised Technical Specifications for operation of the Salem Nuclear Generating Station, Unit No. 1 (the facility) located in Salem County, New Jersey. The amendment is effective as of the date of issuance.

The amendment revises Radiological Safety Technical Specifications related to the Cycle 2 operation of Salem Unit 1.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated March 2, 1979 as supplemented by letters dated April 30, 1979, July 25, 1979, August 3, 1979, August 8, 1979, August 9, 1979 and September 14, 1979, (2) Amendment No. 20 to License No. DPR-70, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Salem Free Public Library, 112 West Broadway, Salem, New Jersey. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 30th day of October, 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

SEPTEMBER 1 2 1979

Docket No. 50-272

REGULATORY DOCKET FILE COPY

Mr. F. P. Librizzi, General Manager
Electric Production
Public Service Electric and Gas Company
80 Park Place, Room 7221
Newark, New Jersey 07101

Dear Mr. Librizzi:

The Commission has issued the enclosed Amendment No. 19 to Facility Operating License No. DPR-70 for the Salem Nuclear Generating Station, Unit No. 1. This amendment consists of changes to the Technical Specifications in response to your request dated December 21, 1977 as supplemented on May 9, 1979.

The amendment makes changes that delete the non-radiological Environmental Technical Specifications (ETS), Appendix B to the license, that are duplicated in the 316(b) Plan of Study required by the Environmental Protection Agency (EPA).

The amendment request also included changes to the radiological ETS. These radiological changes are being reviewed separately.

You requested that the aquatic portion of Section 3.1.2 and all of Section 4.1 be deleted and that the requirements of these sections be met through the 316(b) Plan of Study now required by EPA. This would relieve you from being required to conduct two similar, but separate, environmental monitoring programs for the two different agencies.

This is in accord with Federal government policy regarding duplication of regulations as outlined in the Second Memorandum of Understanding between NRC and EPA.

We discussed the 316(b) Plan with EPA and the other affected natural resource agencies prior to its approval by EPA. We find that it fully meets the concerns identified during our environmental review. Quarterly meetings among NRC staff, the licensee, EPA, National Marine Fisheries Service, U. S. Fish and Wildlife Service and the States of Delaware and

7910020018 KTB P2

OFFICE➤						
SURNAME➤						
DATE➤						

New Jersey have been scheduled to insure that all interest agencies are fully informed and actively involved in any changes that may be made to the 316(b) program. Because the NRC will be directly involved in any decisions that are made concerning the monitoring program, our reliance on the 316(b) Plan and the deletion of the monitoring program currently required by the ETS will not affect our ability to assess the environmental impacts of plant operation.

The 316(b) Plan is designed to collect data specifically for impact assessment and in this respect is superior to the monitoring program now required by Appendix B to the ETS. Hence its substitution by reference in the ETS in place of the current monitoring program will not result in increased environmental impact and is therefore acceptable.

This substitution will provide for reporting results of monitoring conducted under the 316(b) Plan to NRC in the annual report. The bases for the proposed new section of the ETS describe NRC participation in the interagency review group and indicate how decisions to change the 316(b) Plan will be made.

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

OFFICE➤						
SURNAME➤						
DATE➤						

Mr. F. P. Librizzi
Public Service Electric and Gas Company - 3 -

A copy of the Notice of Issuance is also enclosed.

Sincerely,

Original Signed By

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 19 to DPR-70
2. Notice of Issuance

cc: w/enclosures
See next page

Distribution

Docket File 50-272	I&E (5)
NRC PDR	B. Jones (4)
Local PDR	B. Scharf (10)
NRR Rdg	D. Brinkman
ORB1 Rdg	B. Harless
D. Eisenhut	C. Miles
B. Grimes	R. Diggs
W. Gammill	H. Denton
T. J. Carter	ACRS (16)
C. Parrish	TERA
W. Ross	J. Buchanan
Attorney, OELD	
R. Vollmer	
M. Grotenhuis	

7228

9/12/79

OFFICE	DOR:ORB1	DOR:ORB1	DOR:ORB1	DOR:AD:ORP	OELD
SURNAME	WRoss:jb	CSParrish	ASchwencer	WPGammill	B. Smith
DATE	9/4/79	9/6/79	9/10/79	9/7/79	9/10/79

NRC FORM 318 (9-76) NRCM 0240 ★ U.S. GOVERNMENT PRINTING OFFICE: 1978 - 265 - 769

cc: Mark J. Wetterhahn, Esquire
Conner, Moore and Corber
Suite 1050
1747 Pennsylvania Avenue, NW
Washington, D. C. 20006

Richard Fryling, Jr., Esquire
Assistant General Solicitor
Public Service Electric and Gas Company
80 Park Place
Newark, New Jersey 07101

Gene Fisher, Bureau of Chief
Bureau of Radiation Protection
380 Scotch Road
Trenton, New Jersey 08628

Mr. Hank Midura, Manager
Salem Nuclear Generating Station
Public Service Electric and Gas Company
80 Park Place
Newark, New Jersey 07101

Salem Free Library
112 West Broadway
Salem, New Jersey 08079

Leif J. Norrholm
U. S. Nuclear Regulatory Commission
Drawer I
Hancocks Bridge, New Jersey 08038

Attorney General
Department of Law and Public Safety
State House Annex
Trenton, New Jersey 08625

Samuel E. Donelson, Mayor
Lower Alloways Creek Township
Municipal Hall
Hancocks Bridge, New Jersey 08038

Richard B. McGlynn, Commissioner
Department of Public Utilities
State of New Jersey
101 Commerce Street
Newark, New Jersey 07102

Deputy Attorney General
State House Annex
State of New Jersey
36 We[illegible] State Street
Tren[illegible]n, New Jersey 08625

Director, Technical Assessment Divisi[illegible]
Office of Radiation Programs (AW-459)
U. S. Environmental Protection Agency
Crystal Mall #2
Arlington, Virginia 20460

U. S. Environmental Protection Agency
Region II Office
ATTN: EIS COORDINATOR
26 Federal Plaza
New York, New York 10007

Mr. E. N. Schwalje, Manager
of Quality Assurance
Public Service Electric and Gas Compa[illegible]
80 Park Place
Newark, New Jersey 07101

Mr. R. L. Mittl, General Manager
Licensing and Environment
Public Service Electric and Gas Company
80 Park Place
Newark, New Jersey 07101

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

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
PHILADELPHIA ELECTRIC COMPANY
DELMARVA POWER AND LIGHT COMPANY
ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-272

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 19
License No. DPR-70

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

   A. The application for amendment by Public Service Electric and Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) dated December 21, 1977 as supplemented May 9, 1979, 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;

   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.

7910020020

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

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 19 , 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: September 12, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 19

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Revise Appendix B as follows:

Remove Pages	Insert Pages
3.1-9	3.1-9
3.1-10	
3.1-11	
3.1-12	
3.1-13	
3.1-14	3.1-14
3.1-15	3.1-15
3.1-16	
3.1-16a	
3.1-16b	
3.1-16c	
3.1-17	
3.1-18	
3.1-19	
3.1-20	
3.1-21	
3.1-23	3.1-23
3.1-25	
4.1-1	4.1-1
4.1-2	4.1-2
4.1-3	
5.6-1	5.6-1

3.1.2 BIOTIC

3.1.2.1 General Ecological Survey

The primary objective of this survey is to determine the effect of plant operation on the ecology and environment of the Delaware River Estuary and environs. The preoperational biological monitoring was initiated in 1968 and monitoring will be continued for 5 years after Unit No. 2 becomes operational. The program shall be discontinued only after approval by NRC staff. These studies will serve as a basis for assessment of the effects of plant operation on the ecology.

Study Plan

The study area includes the Delaware River Estuary and some tributaries within an approximate 10-mile radius of the station. The biological parameters monitored are listed in Table 3.1-2 and the general sampling locations are shown in Figure 3.1-1.

Physiochemical parameters will be monitored in the various sampling programs and will typically include dissolved oxygen, temperature, salinity, pH, and water transparency.

Specification

1. Aquatic Studies

   (deleted)

This page is intentionally left blank.

Pages 3.1-11 through 3.1-13, pages 3.1-16 through 3.1-23 and page 3.2-25 have been deleted by Amendment No. .

2. Terrestrial Studies

Studies of the Terrestrial Environment shall include:

1. Monitoring of nesting by the diamondback terrapin on Sunken Ship Cove Beach and in regions outside the thermal plume.
2. A monthly (weather permitting) bird survey in the area of Artificial Island.
3. Monitoring occurrence and nesting of the osprey and southern bald eagle within a general 5-mile radius of the station.

Reporting Requirements

Reporting levels shall be developed after one year of full power operation of Unit 2. Post-operational data will be related to preoperational norms from which report levels will be established.

Bases

All biological parameters sampled will provide background data for determining the environmental effects of station operation. Results of the operational studies will be compared with preoperational studies by statistical methods. The various sampling locations were selected on the basis of their representative distribution throughout this region. As the data from these sites are analyzed, it will be determined whether additional sites are needed or old sites can be eliminated. The frequency of sampling has been established in much the same manner.

TABLE 3.1-2

SUMMARY OF TERRESTRIAL AND AERIAL SAMPLING PROGRAM

Sample	Method	Sampling Frequency*	Area Sampled Relative to Station (Mile 0)
Terrestrial and Aerial			
Birds	Visual observations	Biweekly to quarterly	Within 3-5 mile radius
Mammals	Visual observations	Biweekly to quarterly	Within 3-5 mile radius

*In the appropriate season.

4.0 SPECIAL SURVEILLANCE AND STUDY ACTIVITIES

4.1 SECTION 316(b) STUDIES

Requirements

The licensee shall provide the results of studies which are conducted to demonstrate continued compliance with Section 316(b) of the Clean Water Act.

Action

The licensee shall submit copies of all reports submitted to EPA on Section 316(b) studies in accordance with the provisions of Subsection 5.6.1 of these ETS.

Bases

The environmental assessments made in the FES-OL of 1973 (Section 5.4) determined that the impacts on the aquatic biota in the Delaware River estuary resulting from continued operation of Units 1 and 2 with once-through cooling were acceptable, but that a monitoring program should be instituted to verify that there would be no significant change in the pertinent ecological parameters considered in the assessments. The licensee implemented a comprehensive monitoring program beginning with commercial operation of Unit 1, as required by NRC through the Appendix B ETS. By letter from Mr. Eckardt C. Beck, Regional Administrator, U. S. Environmental Protection Agency, to Public Service Electric and Gas Company, dated May 25, 1979, the EPA required the licensee to perform a 316(b) study which is similar to, but more extensive in some respects, than the aquatic surveillance study required by the ETS. Members of NRC staff have discussed this plan of study with EPA and other affected natural resource agencies and have provided input to its development.

An interagency review group, composed of staff from NRC, the licensee, EPA Region II, National Marine Fisheries Service, U. S. Fish and Wildlife Service and the States of Delaware and New Jersey will hold quarterly meetings to insure that all interested agencies are fully informed of program results and actively involved in any changes that may be made.

The submittal of results from the programs required by the NPDES permit will allow the staff to follow the consequences of this licensing action and will therefore satisfy needs identified in the FES. The staff will coordinate review of the results with EPA to determine whether subsequent regulatory action is required.

5.6 PLANT REPORTING REQUIREMENTS

5.6.1 ROUTINE REPORTS

5.6.1.1 Annual Environmental Operating Report

1.a. Nonradiological Report

A report on the environmental surveillance program for the previous 12 months of operation shall be submitted to the Director of Regional Inspection and Enforcement Office (with copy to the Director, Office of Nuclear Reactor Regulation) as a separate document within 90 days after January 1 of each year. The period of the first report shall begin with the date of initial criticality. The report shall include summaries, interpretations, and statistical evaluation of the results of the non-radiological environmental surveillance activities (Section 3.0) and the environmental monitoring programs required by limiting conditions for operation (Section 2.0) for the report period, including a comparison with preoperational studies, operation controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. If harmful effects or evidence of irreversible damage are detected by the monitoring, the licensee shall provide an analysis of the problem and a proposed course of action to alleviate the problem.

b. Reports to Other Agencies

Copies of routine reports required by Federal, State, local, and regional authorities for the protection of the environment shall be submitted to the Director, Office of Nuclear Reactor Regulation, USNRC, for information. Reports of the EPA 316(b) study shall be submitted to the NRC.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-272
PUBLIC SERVICE ELECTRIC AND GAS COMPANY,
PHILADELPHIA ELECTRIC COMPANY,
DELMARVA POWER AND LIGHT COMPANY, AND
ATLANTIC CITY ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 19 to Facility Operating License No. DPR-70, issued to Public Service Electric and Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees), which revised Technical Specifications for operation of the Salem Nuclear Generating Station, Unit No. 1 (the facility) located in Salem County, New Jersey. The amendment is effective as of the date of issuance.

The amendment makes changes that delete those non-radiological Technical Specifications in Appendix B to the License that are duplicated in the 316(b) Plan of Study required by the Environmental Protection Agency.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated December 21, 1977 as supplemented May 9, 1979 and (2) Amendment No. 19 to License No. DPR-70. These items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Salem Free Public Library, 112 West Broadway, Salem, New Jersey. A copy of item (2) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 12th day of September, 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors