Public Service
Electric and Gas
Company

Corbin A. McNeill, Jr.
Vice President -
Nuclear

Public Service Electric and Gas Company P.O. Box 236, Hancocks Bridge, NJ 08038 609 339-4800

August 6, 1985

Ref: LCR 85-09

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Licensing
Washington, D. C. 20555

Attention: Mr. Steven A. Varga, Chief
Operating Reactors Branch 1
Division of Licensing

Gentlemen:

REQUEST FOR AMENDMENT
FACILITY OPERATING LICENSE DPR-70
UNIT NO. 1
SALEM GENERATING STATION
DOCKET NO. 50-272

In accordance with the Atomic Energy Act of 1954, as amended and the regulations thereunder, we hereby transmit copies of our request for amendment and our analyses of the changes to Facility Operating License DPR-70 for Salem Generating Station, Unit No.1.

This amendment request consists of changes to those sections of the Technical Specifications, and to the Facility Operating License, to accommodate an increase in RATED THERMAL POWER. This change will result in identical power ratings for both Salem Units.

In accordance with the fee requirements of 10CFR170.21. a check in the amount of $150.00 is enclosed.

A001 w/check $150.00
#01922541
3/39

4 Corrected Pages
Rec'd 9/4/85

Pursuant to the requirements of 10CRF50.91, a copy of this request for amendment has been sent to the State of New Jersey as indicated below.

This submittal includes three (3) signed originals and forty (40) copies.

Sincerely,

Enclosure

C Mr. Donald C. Fischer
Licensing Project Manager

Mr. Thomas J. Kenny
Senior Resident Inspector

Mr. Samuel J. Collins, Chief
Projects Branch No. 2, DPRP
Region 1

Mr. Frank Cosolito, Acting Chief
Bureau of Radiation Protection
Department of Environmental Protection
380 Scotch Road
Trenton, N.J. 08628

Honorable Charles M. Oberly, III
Attorney General of the State of Delaware
Department of Justice
820 North French Street
Wilmington, Delaware 19801

Ref: LCR 85-09

STATE OF NEW JERSEY)
) SS.
COUNTY OF SALEM)

Corbin A. McNeill, Jr., being duly sworn according to law deposes and says:

I am a Vice President of Public Service Electric and Gas Company, and as such, I find the matters set forth in our letter dated August 6, 1985, concerning our Request for Amendment to Facility Operating License DPR-70, are true to the best of my knowledge, information and belief.

Subscribed and Sworn to before me
this 6TH day of AUGUST, 1985

Rudolph L. von Fischer Jr.
Notary Public of New Jersey

L. von FISCHER JR.
Notary Public of New Jersey
My Commission Expires Sept. 10, 1986

My Commission expires on ____________________

PROPOSED CHANGE TO
TECHNICAL SPECIFICATIONS
SALEM UNIT NO. 1

Ref: LCR 85-09

Description of Change:

Increase the Unit 1 licensed core power from 3338 MWT to 3411 MWT. The necessary Tech. Spec. changes are as follows:

a. Section 1.25, change RATED THERMAL POWER from 3338 to 3411 MWT

b. Section 2.2, change RTS setpoints for core flow from 88,500 gpm/loop to 87,300 gpm/loop

c. Section 3.2.5, change DNB parameters for RCS Tavg from 581°F to 582°F

The changes to Section 1.25 and 3.2.5 are a direct result of the power uprate.

The change to Section 2.2 for core flow is for consistency with the core flow requirements of Section 3.2.5. Section 3.2.5 requires a flow of 349,200 gpm total for four loops (which equals 87,300 gpm per loop). The value of 87,300 gpm is consistent with Unit 2 Technical Specifications and with the design flow used by Westinghouse for both plants.

Reason for Change:

Unit 1 and Unit 2 are essentially identical units. This change will allow Unit 1 to operate at the same power as Unit 2 which is licensed for a core power of 3411 MWT. Operating the units at the same power will allow greater standardization between the units and an increase in electrical output from Unit 1.

Significant Hazards Consideration Evaluation

The attached evaluation documents the review done by PSE&G and Westinghouse to establish that operating Unit 1 at a core power of 3411 MWT does not represent a significant hazards consideration. These proposed changes may cause some increase in the consequences of a previously analyzed

Significant Hazards Consideration Evaluation (Cont.)

accident or some slight decrease in a margin of safety; but, the results of the change on plant operation will remain clearly within the bounds of the FSAR analyses and within the guidelines of the Standard Review Plan Sections 4.3 and 4.4.

These changes also bring about consistency between the Technical Specifications for both Salem Units; the changes, therefore, correspond to examples (vi) and (i) of the guidance provided by the Commission on changes considered "Not Likely to Involve A Significant Hazards Consideration" in Federal Register 48FR14870.

PHYSICS TESTS

1.20 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the Updated FSAR, 2) authorized under the provisions of 10CFR50.59, or 3) otherwise by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.21 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

1.22 The PROCESS CONTROL PROGRAM shall be that program which contains the current formula, sampling, analyses, test, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes, based on demonstrated processing of actual or simulated wet solid wastes, will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 71 and Federal and State regulations and other requirements governing the disposal of the radioactive waste.

PURGE - PURGING

1.23 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

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

*Design flow is 87,300 gpm per loop.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 4 Loops In Operation	LIMITS 3 Loops in Operation
Reactor Coolant System T_{avg}	$\leq$ 582°F	$\leq$ 572°F
Pressurizer Pressure	$\geq$ 2220 psia*	$\geq$ 2220 psia*
Reactor Coolant System	$\geq$ 349,200 gpm	$\geq$ 278,100 gpm

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

PSEG

Public Service Electric and Gas Company P.O. Box 236 Hancocks Bridge, New Jersey 08038

Nuclear Department

TITLE: NSSS POWER UPRATING TO 3423 MWt
SALEM UNIT 1
SALEM NUCLEAR GENERATING STATION

PURPOSE:

The purpose of this Evaluation is to demonstrate that the proposed changes to technical Specifications to accommodate an uprate in core power for Salem Unit 1 from 3338 MWt to 3411 MWt does not involve a significant hazards consideration.

SCOPE:

This proposed power uprate is for Unit 1 only; however, the proposed power for Unit 1 is the present licensed power of Unit 2. The power capability parameters are listed in Table 1 of this safety evaluation. A core power of 3411 MWt corresponds to a NSSS power of 3423 MWt. The difference, 12 MW, is due to RCP pump heat.

This evaluation covers both the Nuclear Steam Supply System (NSSS) and the Balance-of-Plant (BOP) design.

ATTACHMENTS:

#1 Westinghouse Report "Salem Unit 1, 3423 MWt NSSS Uprating, Safety Evaluation", transmitted by PSE-85-552 dated April 2, 1985.

#2 Draft FSAR Changes To Be Incorporated Upon Issuance Of The Salem Unit 1 Power Uprate Amendment

GENERAL DISCUSSION:

The primary and secondary plant operating parameters selected for the Unit 1 uprate are the existing parameters for Unit 2. The rationale for selecting these specific values is two fold. First, as will be demonstrated later in this section of the report, the units are essentially identical (Salem Units 1 and 2 form a twin unit station). Nearly all of the original equipment criteria and analyses are common to both Units, and these common power related criteria and analyses assumed the Unit 2 higher power. Secondly, uprating Unit 1 to Unit 2's power will allow even greater standardization between the Units. The selected primary and secondary plant operations parameters are listed in Table 1.

The uprate was demonstrated by verifying that at the uprated condition, all current FSAR analyses remain applicable and by verifying that all equipment and systems for Unit 1 remain within their current specifications, acceptance criteria, and qualifications. In some cases, the documentation for equipment had to be revised to reflect the uprated operating conditions.

Feasibility had earlier been demonstrated by verifying that all power related equipment in Unit 1 was identical or functionally identical to Unit 2 equipment.

A fundamental basis of this uprate evaluation is that the present licensing criteria and acceptance standards for current Unit 1 operations remain applicable to the uprated Unit 1. All equipment reviews and evaluations were likewise done to the current Unit 1 codes and standards.

The impact of the uprated condition on the actual Unit 1 operating temperatures, pressures, and flows is small. These changes in actual operating conditions do not cause any of the systems to exceed it's design parameters.

The RCS T-hot increases 1.7° F; T-average increases 1.1° F; and T-cold increases 0.6° F. The CVCS letdown line temperature correspondingly increases 0.6° F. There is no increase in the RCS operating pressure and flow.

The main steam temperature and pressure do not change. The feedwater temperature at the inlet of the Steam Generator increases by 3.8° F. Feedwater pressure does not change. Main steam and feedwater flows increase by 2.7%. Other systems connected to these systems, e.g. condensate, turbine drains, will also see similar increases in flow and minor changes in temperature.

The flows and pressures in the auxiliary cooling systems - component cooling water, service water, and circulating water - do not increase. The heat load increase to the component cooling water and service water is very marginal; the heat load increase to the circulating water results in a 0.3° F increase in the return temperature to the river.

As part of the overall evaluation, Westinghouse was authorized to reevaluate at the uprated condition all the equipment, systems, and accident events which Westinghouse performed during the original plant design. The Westinghouse report, Attachment 1, is in summary form incorporated into this evaluation. The results of the review are diccussed below.

Impact on Accident Analyses

All currently docketed FSAR accident analyses were reviewed. These include Loss of Coolant Accident (LOCA) events, High Energy Pipe

Breaks Outside Containment, Containment Analyses, Subcompartment Pressure Analyses, Main Steam Line Breaks, and other non-LOCA events. All the current accident analyses either originally assumed Unit 2's power (or higher), or else they are insensitive to the power increase. None of the analyses will need to be modified to implement the Unit 1 uprate.

The discussion below on accidents supplements the discussion in Attachment 1, Section 3.

High Energy Break Analysis - The power uprate does not change the pressure in any line and has only a small temperature increase in a few lines. Since there is no pressure increase in any line, there is no change to the pipe whip or jet impingement loads. The changes to the service temperature of high energy lines analyzed for a break outside containment are limited to:

(1) Feedwater (penetreation area) +0.3° F

(2) CVCS Letdown (betwen the Regenerative heat exchanger and the Letdown heat exchanger) +0.3° F

The increase in feedwater temperature has no impact on the Unit 1 High Energy Break Analysis. A comparison of FSAR Section 3.6.5.2 and 3.6.5.3 shows that in all cases main steam line breaks are postulated in the same safety related areas where feedwater line breaks are postulated. The main steam line breaks remain limiting. In fact, there is only one short run (approximately 15 feet) of feedwater piping in a safety area where there is not also main steam piping. Refer to FSAR Figures 3.6-11 and 3.6-15. However, due to the low stress values, there are no postulated breaks in this section of feedwater piping.

The increase in the CVCS letdown temperature, 0.3° F, is considered insignificant and well within the accuracy of existing calculations and conservatism.

Subcompartment Pressure Analysis - The Salem subcompartment pressure analysis uses common assumptions which envelope both units; therefore, the report (Reference 64 of SNGS UFSAR Section 15.4) remains applicable for Unit 1 at the uprated condition.

Containment Analyses - The containment transient analyses presented in the SNGS FSAR Section 15.4.8 are common to both units. The transient analyses for RCS pipe breaks assumed a NSSS power of 102% of 3570 MWt. The transient analyses for main steam line breaks assumed a NSSS power of 102% of 3425 MWt.

The radiological consequences of the postulated design bases loss-of-coolant-accident assumed a reactor power of 3558 MWt. The hydrogen production and accumulation analysis assumed a reactor power of 3575 MWt.

Impact to Primary Plant Components

The original equipment Specifications for the Unit 1 Reactor Coolant Pumps (RCP), Control Rod Drive Mechanism (CRDM), Reactor Coolant Piping, Pressurizer, and Steam Generators enveloped the proposed power uprate and only minor documentation changes were required to reflect the proposed, new operating conditions.

The Unit 1 Reactor Vessel and some of its internals were originally analyzed for 3338 MWt power. These analyses were revised for the uprated condition.

During the initial feasibility study it was determined that all Unit 1 primary components are essentially identical to the Unit 2 components except for the following:

1) The flow hole patterns in the lower reactor internals baffle barrel region formers are different, and

2) The control rod patterns are different.

The flow hole and control rod patterns were evaluated as having no impact on the capability to uprate. (As part of a standardization task separate from the Unit 1 uprate, the control rod pattern on Unit 2 was modified starting with cycle 3 to more closely resemble Unit 1's pattern).

A third difference, drainage capacity from the moisture separators in the Steam Generators, was identified during the feasibility review. However, during Unit 1's fifth refueling outage, the drainage capacity was increased to the same capacity as Unit 2's. This modification was necessary to ensure that moisture carryover from the Steam Generators is kept below 0.25% when Unit 1 is operated at 3423 MWt. This was the only hardware modification required for the Unit 1 NSSS power uprate.

Since the original End-of-Life (EOL) neutron fluence for Unit 1's Reactor Vessel was based on a core power of 3483 MWt, a power higher than the proposed uprated core power, and because Salem 1 has shifted to a low neutron flux leakage core starting with Cycle 6, the original EOL neutron fluence calculation remains bounding.

Impact to Piping

The RCS piping and all other Nuclear Class 1 piping six inches in diameter or greater was originally analyzed by Westinghouse. Westinghouse in Attachment 1 reviewed the piping analyses and determined that the original analyses still envelope the uprated condition.

PSE&G reviewed the class 1 piping below six inches in diameter. This review indicated that the transients used for PSE&G analyzed Class 1 piping likewise enveloped the proposed power uprate.

All Nuclear Class 2 and 3 piping and all non-nuclear piping was analyzed by PSE&G using criteria that enveloped Salem Units 1 and 2. No reevaluation is necessary to any of these analyses.

The pressurizer safety and relief valve discharge piping on Unit 1 was analyzed for the relief valve's rated flow condition. The safety and relief valves are identical between the two units.

Impact to Reactor Fluid Systems

The following system design documentation was reviewed:

Reactor Coolant
Safety Injection
Chemical and Volume Control
Residual Heat Removal
Containment Spray
Spent Fuel Pool Cooling
Sampling

It was determined that these systems were originally designed using common (Unit 2) power parameters.

Impact to Reactor Auxiliary Equipment

All power related Reactor Auxiliary Equipment was reviewed. All Unit 1 equipment was found to be functionally or physically identical to the Unit 2 equipment except for the seal water injection filters. The filter capacity of Unit 1 is 300 gpm and for Unit 2 it is 350 gpm, both of which are more than adequate for handling the seal water flow rate. The equipment reviewed includes valves, pumps, heat exchangers, tanks, demineralizers, and filters. Accordingly, all the auxiliary equipment is considered capable of performing satisfactorily at the uprated power of 3423 MWt.

NSSS/BOP Interfaces - The original Balance-of-Plant interfaces specified by Westinghouse are identical for both units and were based on Unit 2's higher power rating. Accordingly, any BOP equipment/component affected by the Unit 1 uprate should remain within its original design criteria.

Impact to Secondary Plant Components and Systems

There are minor differences between the Unit 1 and 2 turbine, but as a result of modifications done during earlier outages, the Unit 1 turbine can operate at loads in excess of the expected load at the uprated condition. At the uprated conditions, the turbine will remain below the original maximum calculated load. The turbine will not require any modification.

The Westinghouse generator on Unit 1 and the General Electric generator on Unit 2 are of equal rating.

The rest of the secondary plant (i.e., main steam, auxiliary feedwater, condensate, feedwater, turbine drains, steam dumps, circulating water, turbine service water, etc.) is considered identical. Based on a review of the PSE&G purchase specifications, all power related equipment purchased directly by PSE&G was determined to be identical between the two units. The equipment for the two units was bought at the same time, from the same vendors, to the same criteria using purchase specifications common to both units. The sole exception is the purchase specification for the main steam isolation valves which specifies slightly higher flow rates for the Unit 2 valves than for the Unit 1 valves. The valves supplied by the vendor are, however, identical for both units. The criteria of the common purchase specification were based on the higher requirements of Unit 2.

One modification, upgrade of the condensate pumps, has been identified as a desirable modification for plant reliability. This modification will decrease feedwater pump low suction pressure trips during secondary plant transients. The modification has been authorized for both units. The Unit 1 modification is scheduled to be implemented during its sixth refueling outage. The modification was done on Unit 2 during its second refueling outage.

Impact to Electrical Plant

Since all electrically operated equipment in Unit 1 was originally specified for the higher power rating of Unit 2, no changes are required to any motor, cable, or switchgear. There will be no increase in the calculated loading to the emergency generators. There will be only a negligible increase in the normal electrical requirements for Unit 1 (The condensate pump upgrade will impact the electrical plant, but that evaluation was done as part of the pump upgrade). The electrical system from the generator to the grid can handle the increased electrical output estimated at 26 MWe.

Impact to Controls and Instrumentation

There is no impact on the controls and instrumentation other than a number of setpoints and calibrations will have to be done for the higher power. A review indicated that recalibration will be required on the Nuclear Instrumentation System and on the Feedwater and Main Steam flow instrumentation and controls.

Impact to Equipment Environmental Qualifications

All equipment environmental qualifications are common to both units, and these assumed the higher power level of Unit 2. No documentation changes were necessary in this area for the Unit 1 power uprate.

Impact to Nuclear Fuels

The Unit 1 Cycle 6 Reload Safety Evaluation (RSE) was performed assuming the present power limit of 3338 MWt; however, a preliminary review indicates that the power uprate would not have a significant impact on the core. The cycle 7 core will be designed for 3411 MWt. The RSE will address the core's safety parameters.

Environmental Impact

The Unit 1 uprate will not cause the Salem Nuclear Generating Station to exceed its allowable heat discharge rate to the river (16.3×10^9 BTU/hr, total both units) nor the maximum allowable difference between river intake and discharge temperatures (27.5° F).

No revision is required to the source term for the radioactive waste management (FSAR, Chapter 11) and for radiation protection (Chapter 12) since it is based on a 3558 MWt core power. The uprate should not have an impact on the volume of radioactive wastes because these are dependent on maintenance and operating events. The slight power change will not in any significant manner alter maintenance and operating events.

Licensing Changes

The FSAR changes identified are listed in Attachment 2. These changes include those identified by Westinghouse in Attachment 1.

The Unit 1 uprate will not cause the SNGS to exceed its environmental (thermal) discharge limits.

If changes to the Technical Specifications for the core physics parameters are required, they will be addressed by a fuel Safety Evaluation as discussed previously. None are forseen.

The Unit 1 Technical Specification changes that have otherwise been identified are limited to:

Date: 6/14/85

Section	Title	Change
1.25	Rated Thermal Power	from 3338 to 3411 MWt
2.2	RTS Setpoints	Core flow from 88,500 gpm/loop to 87,300 gpm/ loop
3.2.5	DNB Parameters	RCS Tavg from 581° F to 582° F

The changes to Section 1.25 and 3.2.5 are a direct result of the power uprate.

The change to Section 2.2 for core flow is for consistency with the core flow requirements of Section 3.2.5. Section 3.2.5 requires a flow of 349,200 gpm total for four loops (which equals 87,300 gpm per loop). The value of 87,300 gpm is consistent with Unit 2 Technical Specifications and with Westinghouse primary plant parameters, Table 1, for both units.

No change will be required in the overtemperature T and overpower T setpoints. The Unit 1 and Unit 2 values are already equal.

CONCLUSION:

The Salem Unit 1 core power uprate from 3338 MWt to 3411 MWt has been thoroughly reviewed by both Westinghouse (Attachment 1) and PSE&G to establish the following:

1. No equipment has to be added or deleted.

2. Except as noted in paragraph 3 below, all systems and equipment affected by the power uprate were either initially specified to conditions which bound the proposed power uprate, or it has been determined by revising the documentation to reflect the proposed power uprate, that the systems and equipment will continue to meet all their current specifications, acceptance criteria, and qualifications.

3. As a result of the Unit 1 Steam Generator moisture separator drain modifications done during the fifth refueling outage, all power related NSSS equipment between Unit 1 and Unit 2 is either identical or functionally identical. Similarly, after the Unit 1 condensate pumps are upgraded during Unit 1's sixth refueling outage, all power related BOP equipment between the units will again be either identical or functionally identical (the Unit 2 pumps were upgraded during Unit 2's second refueling outage. This change is a plant improvement project which will increase the reliability of normal feedwater to

steam generators. It is not a requirement for meeting the original design bases).

4. The functions and requirements for all systems and equipment remain unchanged.

5. None of the accident analyses for Salem 1 have to be modified.

6. The licensing bases and environmental impact for the unit as defined by the FSAR and the Technical Specifications are not altered.

In addition further assurances of safe operation of Salem Unit 1 at 3411 MWt have been provided by the operation of Salem Unit 2 at 3411 MWt. Accordingly, the proposed uprate in Unit 1 core power starting with fuel cycle 7 does not represent a significant hazards consideration.

TABLE 1

SALEM UNIT 1 POWER CAPABILITY PARAMETERS

	Present Power	Uprated Power*
NSSS Power, MWt	3350	3423
Core Power, MWt	3338	3411
Thermal Design Flow, Loop gpm	87,300	87,300
Reactor Flow, Total, 10^6 lbm/hr	132.3	132.3
Reactor Coolant Pressure, psia	2250	2250
Reactor Coolant Temperature, °F		
Core Outlet	611.8	613.7
Vessel Outlet	609.1	610.8
Core Average	579.8	581.0
Vessel Average	576.8	577.9
Vessel/Core inlet	544.4	545.0
Steam Generator Outlet	544.2	544.8
Steam Generator		
Steam Temperature, °F	519.0	519.0
Steam Pressure, psia	805	805
Steam Flow, 10^6 lbm/hr Total	14.47	14.86
Feedwater Temperature, °F	429.0	432.8
Zero Load Temperature, °F	547	547
Percent Tube Plugging	0	0
Core Bypass Percent	4.5	4.5
Fuel Design	17 x 17 STD	17 x 17 STD
Gross Electrical Output, MWe	1132	1158

*These parameters are identical to the Unit 2 values.