ATTACHMENT # 1

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PUBLIC SERVICE ELECTRIC AND GAS COMPANY
SALEM UNIT 1
NSSS UPRATING - 3423 MWt
SAFETY EVALUATION REPORT

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OBJECTIVES

Public Service Electric and Gas Company (PSE&G) is applying to the Nuclear Regulatory Commission for approval to operate Salem Unit 1 at a licensed power rating of 3423 MWt. Salem Unit 1 is currently licensed to operate at 3350 MWt. PSE&G has authorized Westinghouse to perform a safety evaluation of NSSS designs, operations, and analyses to provide the following information relevant to that application:

1. A description of the proposed change in the licensed power rating of Salem Unit 1.

2. An assessment of the impact of that change on NSSS equipment designs, safety analyses, and systems operations.

3. Technical information to be used by PSE&G in support of its application for the increase in licensed power rating.

4. A technical basis for establishing that the proposed increase in power rating does not involve an unreviewed safety question in accordance with requirements of 10 CFR 50.59.

This report summarizes the results of the safety evaluation performed by Westinghouse, and presents the conclusions based upon it.

CONCLUSIONS

The proposed increase in the licensed power rating of Salem Unit 1 has been reviewed in detail with respect to its impact on the following aspects of NSSS design and operation:

1. The consequences of accidents postulated in the FSAR.

2. The capability of systems and equipment to meet design bases specified in the FSAR.

3. The capability of equipment to maintain structural integrity under conditions defined in the FSAR.

4. Definition of NSSS/BOP safety related interfaces.

5. Operating limits and conditions contained in Technical Specifications that are impacted by the power rating increase.

This review has demonstrated that Salem Unit 1 is capable, in its present design configuration, of operating at the proposed power rating within the compliance specifications of the design criteria or safety limits contained in the FSAR for NSSS systems and equipment, providing that the plant is operated in accordance with the Technical Specification changes proposed by Westinghouse. The review has verified that:

1. The probability of a malfunction of NSSS equipment important to safety previously evaluated in the FSAR will not be increased at the proposed power rating.

2. The consequences of a malfunction of NSSS equipment important to safety previously evaluated in the FSAR will not be increased at the proposed power rating.

3. The possibility of a malfunction of NSSS equipment important to safety different from any already evaluated in the FSAR is not created by operation at the proposed power rating.

4. The margin of safety as defined in the bases to any technical specification will not be reduced by operation at the proposed power rating.

Therefore, it has been concluded that operation of Salem Unit 1 at the increased power rating does not reduce the NSSS safety margins, and does not involve an unreviewed question as defined by 10 CFR 50.59.

Customer Reference No(s).
PO #928644

Westinghouse Reference No(s).
(Change Control or RFQ As Applicable)
GO #NA42059

WESTINGHOUSE

NUCLEAR SAFETY EVALUATION CHECK LIST

1) NUCLEAR PLANT(S) Salem Unit 1

2) CHECK LIST APPLICABLE TO: 2% Power Uprating
(Subject of Change)

3) The written safety evaluation of the revised procedure, design change or modification required by 10CFR50.59 has been prepared to the extent required and is attached. If a safety evaluation is not required or is incomplete for any reason, explain on Page

Parts A and B of this Safety Evaluation Check List are to be completed only on the basis of the safety evaluation performed.

CHECK LIST - PART A

(3.1) Yes ___ No _X_ A change to the plant as described in the FSAR?

(3.2) Yes ___ No _X_ A change to procedures as described in the FSAR?

(3.3) Yes ___ No _X_ A test or experiment not described in the FSAR?

(3.4) Yes _X_ No ___ A change to the plant technical specifications (Appendix A to the Operating License)?

4) CHECK LIST - PART B (Justification for Part B answers must be included on Page 2.)

(4.1) Yes ___ No _X_ Will the probability of an accident previously evaluated in the FSAR be increased?

(4.2) Yes ___ No _X_ Will the consequences of an accident previously evaluated in the FSAR be increased?

(4.3) Yes ___ No _X_ May the possibility of an accident which is different than any already evaluated in the FSAR be created?

(4.4) Yes ___ No _X_ Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

(4.5) Yes ___ No _X_ Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

(4.6) Yes ___ No _X_ May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?

(4.7) Yes ___ No _X_ Will the margin of safety as defined in the bases to any technical specification be reduced?

If the answers to any of the above questions are unknown, indicate under 5) REMARKS and explain below.

If the answer to any of the above questions in 4) cannot be answered in the negative based on written safety evaluation, the change cannot be approved without an application for license amendment submitted to NRC pursuant to 10CFR50.90.

5) REMARKS:

None

The following summarizes the justification upon the written safety evaluation,(1) for answers given in Part B of the Safety Evaluation Check List:

The proposed 2% uprating of Salem Unit 1 from 3350 MWt to 3423 MWt has been reviewed in detail This uprating will license Unit 1 to operate at the same power level as Unit 2, which has been operating safely at a power output of 3423 MWt. The Unit 1 uprating has been thoroughly evaluated in the areas of accident analyses, NSSS Systems, NSSS components, NSSS/BOP interface and FSAR and Technical Specification impact. Based on this review, it is concluded that a 2% uprating is acceptable and involves no unreviewed safety question. Salem Unit 1 can be safely operated at a power output of 3423 MWt without undue risk to public health and safety.

(1) Reference to document(s) containing written safety evaluation: Salem Unit 1 NSSS Uprating 3423 MWt Safety Evaluation

FOR FSAR UPDATE

Section: ________ Page(s): ________ Table(s): ________ Figure(s): ________

Reason for/Description of Change:

Prepared by (Nuclear Safety): R H Ous Date: 3-22-85

Coordinated with Engineer(s): Gerald B Smith Date: 3-25-85

rdinating Group Manager(s): R. A. McStoturtge Date: 3-25-85

Nuclear Safety Group Manager: [illegible] Date: 3/25/85

SECTION 1
INTRODUCTION

Public Service Electric and Gas Company (PSE&G) is engaged in a program to increase the electrical output of Salem Unit 1. The program is directed toward obtaining approval from the USNRC to operate the plant at a slightly increased power level. At present, Salem Unit 1 is licensed to operate at an NSSS power rating of 3350 MWt. PSE&G is applying for an amendment to the operating license that will permit operation at 3423 MWt, an increase of 2.2%.

As a part of the program to uprate Salem Unit 1, PSE&G authorized Westinghouse to perform a review of NSSS systems and equipment designs to verify their capability to meet requirements for operation at 3423 MWt. That review was conducted in accordance with groundrules and criteria put forth in Westinghouse topical report WCAP-10263, "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant". A summary of the major guidelines followed in the NSSS design review follow:

1. Scope of Review

   The review encompassed all aspects of the Salem Unit 1 NSSS design and operation that were impacted by the power increase.

2. Safety Review Acceptance Criteria

   NSSS designs have been reviewed to verify compliance at the increased power rating with licensing criteria and standards currently required by the Salem Unit 1 operating license. In addition, a review has been made as defined in 10 CFR 50.59 to identify any potential unreviewed safety question that might occur as a result of the increased power rating.

3. Structural Review Acceptance Criteria

The structural design of NSSS equipment was reviewed to assure that compliance has been maintained at the increased power rating with industry codes and standards that applied when the equipment was originally built.

4. Functional Capability

A review has been made to verify that NSSS components and systems will continue to meet functional requirements specified in the FSAR at the increased power rating.

5. Analytical Techniques

Current NRC approved analytical techniques have been used for analyses performed at the increased power rating.

6. Balance of Plant Interfaces

Information provided by Westinghouse to other design groups has been reviewed and revised when impacted by the increase in power rating.

2.0 OPERATING PARAMETERS

2.1 ORIGINAL DESIGN PARAMETERS

Salem Unit 1 was originally designed to operate at a guaranteed NSSS power rating of 3350 MWt with a steam pressure of 805 psia. NSSS operating parameters for these conditions are listed in Table 2-1.

2.2 UPRATED OPERATING PARAMETERS

At uprated power conditions, Salem Unit 1 will operate at 3423 MWt with 805 psia steam pressure. Table 2-1 lists primary plant parameters for the uprated power operating conditions. Note that the uprated Salem Unit 1 parameters are identical to that of Salem Unit 2.

2.3 COMPARISON OF PARAMETERS

A comparison of the operating parameters listed in Table 2-1 shows that operating conditions at the uprated power are identical to the Salem Unit 2 design conditions. A detailed comparison of the more significant parameters indicates the following:

1. NSSS Power

   The uprating will increase the NSSS thermal power by about 2.2% when compared to the Salem 1 original design conditions.

2. Reactor Flow

   Thermal design calculations at the current power of 3350 MWt are based on a reactor inlet flow of 349,200 gpm. Thermal design calculations for the uprated conditions were based on the same reactor inlet flow of 349,200 gpm. This is the same thermal design flowrate used for Salem Unit 2.

3. Reactor Coolant Temperatures

As shown by Table 2-1, reactor coolant temperatures for the uprated conditions do not differ significantly from those for the current 3350 MWt power level. As would be expected, the higher power level is reflected by a slightly greater temperature rise in the coolant as it passes through the reactor vessel. Figure 2-1 provides a graphical comparison of the reactor vessel cold leg, hot leg and vessel average temperatures. It shows that there is little difference between these parameters for the current and uprated operating modes throughout the power range.

4. Steam Pressure

The steam pressure is being maintained at 805 psia and is identical to the steam pressure of Salem Unit 2.

5. Steam Flow

Steam flow at the 3423 MWt conditions has increased over the 3350 MWt condition roughly in proportion to the thermal power increase of 2.2 percent.

2.4 CONCLUSIONS

Comparison of the data presented in Table 2-1 and in Figure 2-1 leads to the conclusions that conditions proposed for future 3423 MWt operation are:

- Not significantly different from those for the original plant design at 3350 MWt with 805 psia steam pressure.
- Identical to currently licensed operating conditions for Salem Unit 2.

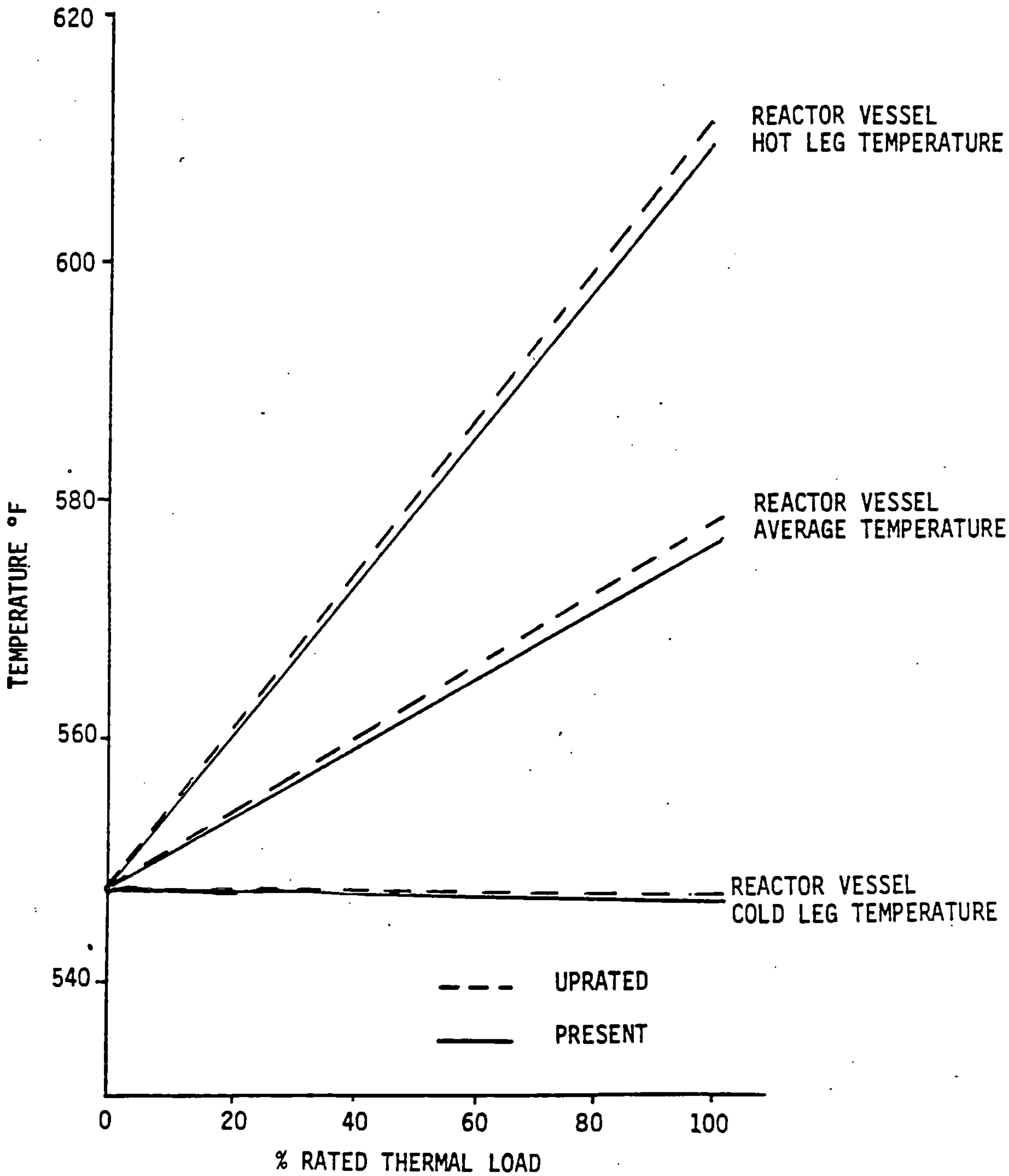
TABLE 2-1

COMPARISON OF REACTOR COOLANT SYSTEM PARAMETERS

	Salem Unit 1 Design Conditions		Salem Unit 2 Design Conditions
	Current License	Uprated License	Current License
NSSS Power, MWt	3350	3423	3423
Reactor Power, MWt	3338	3411	3411
Reactor Coolant Pump Heat, MWt	12	12	12
Reactor Flow, Total, gpm	349,200	349,200	349,200
Reactor Flow, Total, million lbm/hr	132.3	132.2	132.2
Reactor Coolant Pressure, psia	2250	2250	2250
Reactor Coolant Temperature, °F			
Core Outlet	611.8	613.7	613.7
Vessel Outlet	609.1	610.8	610.8
Core Average	579.8	581.0	581.0
Vessel Average	576.8	577.9	577.9
Vessel/Core Inlet	544.4	545.0	545.0
Steam Generator Outlet	544.2	544.8	544.8
Steam Generator			
Steam Temperature, °F	519.0	519.0	519.0
Steam Pressure, psia	805	805	805
Steam Flow, Total, million lbm/hr	14.47	14.86	14.86
Zero Load Temperature, °F	547	547	547
Core Bypass, percent	4.5	4.5	4.5
Fuel Design	17x17	17x17	17x17

Figure 2-1

REACTOR COOLANT TEMPERATURES
VERSUS
PERCENT RATED LOAD



3.0 ACCIDENT ANALYSIS

3.1 INTRODUCTION

The following discussion summarizes the safety analysis evaluation performed to assess the effect of increasing the rated thermal power for Salem Unit 1 from 3350 MWt to 3423 MWt with respect to Salem Generating Station FSAR Chapter 15.

3.2 Classification of Plant Conditions

Since 1970 the American Nuclear Society (ANS) classification of plant conditions has been used which divides plant conditions into four categories in accordance with anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are as follows:

- Condition I: Normal Operation and Operational Transients
- Condition II: Faults of Moderate Frequency
- Condition III: Infrequent Faults
- Condition IV: Limiting Faults

The basic principle applied in relating design requirements to each of the conditions is that the most probable occurrences should yield the least radiological risk to the public and those extreme situations having the potential for the greatest risk to the public shall be those least likely to occur. Where applicable, reactor trip system and engineered safeguards functioning is assumed to the extent allowed by considerations, such as the single failure criterion, in fulfilling this principle.

3.3 INITIAL POWER CONDITIONS ASSUMED IN THE PRESENT SALEM FSAR ACCIDENT ANALYSIS

Table 4-1 lists the principal power rating values which were assumed in analyses performed in the Salem FSAR. Two ratings are given:

1. <u>The guaranteed Nuclear Steam Supply System thermal power output</u>: This power output includes the guaranteed core thermal power generation and the thermal power generated by the reactor coolant pumps.

2. <u>The Engineered Safety Features design rating</u>: The Westinghouse supplied Engineered Safety Features are designed for a thermal power higher than the guaranteed value in order not to preclude realization of future potential power capability. This higher thermal power value is designated as the Engineered Safety Features design rating. This power output includes the thermal power generated by the reactor coolant pumps.

Where initial power operating conditions are assumed in accident analyses, the "guaranteed Nuclear Steam Supply System thermal power output" plus allowance for errors in steady state power determination is assumed. Where demonstration of adequacy of the containment and Engineered Safety Features are concerned, the "Engineered Safety Features design rating" plus allowance for error is assumed. The thermal power values for each transient analyzed are given in Table 3-2.

3.4 CONCLUSIONS

The currently docketed Salem FSAR LOCA and non-LOCA accidents that are not zero power transients were performed at $\geq$ 3423 MWt NSSS power level. Therefore no additional analysis needed be performed. Revisions to the Salem FSAR and technical specifications are required. These page changes are presented in Section 8.

Operation of Salem Unit 1 at the increased power rating of 3423 MWt does not reduce the NSSS safety margins, and does not involve an unreviewed question as defined by 10 CFR 50.59. Salem Unit 1 is capable, in its present design configuration, of operating at 3423 MWt within the compliance specifications of the design criteria or safety limits specified in the Salem FSAR.

TABLE 3-1*

NUCLEAR STEAM SUPPLY SYSTEM POWER RATINGS

Guaranteed Nuclear Steam Supply System thermal power output	3423 MWt
The Engineered Safety Features design rating (maximum calculated turbine rating)	3577 MWt
Thermal power generated by the reactor coolant pumps	12 MWt
Guaranteed Core Thermal Power	3411 MWt

*Table 15.1-1 from Salem FSAR.

TABLE 3.2* (Sheet 1 of 4)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

FAULTS	COMPUTER CODES USED	REACTIVITY COEFFICIENTS ASSUMED: MODERATOR[1] TEMPERATURE ($\Delta k/^{\circ}F$)	MODERATOR[1] DENSITY (ΔK/gm-cc)	DOPPLER[2]	INITIAL NSSS THERMAL POWER OUTPUT ASSUMED (MWt)
CONDITION II					
Uncontrolled RCC assembly Bank Withdrawal from a Subcritical Condition	WIT-6, FACTRAN	$+1 \times 10^{-5}$	---	Lower	0
Uncontrolled RCC Assembly Bank Withdrawal at Power	LOFTRAN	---	0 and 0.43	lower and upper	3423
RCC Assembly Misalignment	THINC, TURTLE, LOFTRAN	---	0	upper	3423
Uncontrolled Boron Dilution	NA	NA	NA	NA	0 and 3423
Partial Loss of Forced Reactor Coolant Flow	PHOENIX, LOFTRAN THINC, FACTRAN	---	0	upper	2396 and 3423
Start-up of an Inactive Reactor Coolant Loop	MARVEL, THINC	---	0.43	lower	2369
Loss of External Electrical Load and/or Turbine Trip	LOFTRAN	---	0 and 0.43	upper	3423
Loss of Normal Feedwater	BLKOUT	---	NA	NA	3577
Loss of Off-Site Power to the Plant Auxiliaries (Plant Blackout)	BLKOUT	---	NA	NA	3423

*Table 15.1-2 of Salem FSAR

TABLE 3.2 (Sheet 2 of 4)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

FAULTS	COMPUTER CODES USED	REACTIVITY COEFFICIENTS ASSUMED: MODERATOR[1] TEMPERATURE (Δk/°F)	MODERATOR[1] DENSITY (Δk/gm-cc)	DOPPLER[2]	INITIAL NSSS THERMAL POWER OUTPUT ASSUMED (MWt)
CONDITION II (Cont'd.)					
Excessive Heat Removal Due to Feedwater System Malfunctions	MARVEL	---	0.43	lower	0 and 3423
Excessive Load Increase	LOFTRAN	---	0 and 0.43	lower	3423
Accidental Depressurization of the Reactor Coolant System	LOFTRAN	---	0	upper	3423
Accidental Depressurization of the Main Steam System	MARVEL	---	Function of Moderator Density See Sec. 15.2.13 (Fig. 15.2.41) Salem FSAR	-2.2 pcm/PF	0 (Subcritical)
Inadvertent Operation of ECCS During Power Operation	LOFTRAN	---	0	lower	3423
CONDITION III					
Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipe which Actuate Emergency Core Cooling	WFLASH, LOCTA-R2				3577

TABLE 3.2 (Sheet 3 of 4)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

FAULTS	COMPUTER CODES USED	REACTIVITY COEFFICIENTS ASSUMED: MODERATOR(1) TEMPERATURE (Δk/°F)	MODERATOR(1) DENSITY (ΔK/gm-cc)	DOPPLER(2)	INITIAL NSSS THERMAL POWER OUTPUT ASSUMED (MWt)
CONDITION III (Cont'd.)					
Inadvertent Loading of a Fuel Assembly into an Improper Position	LEOPARD, TURTLE	---	NA	NA	3423
Complete Loss of Forced Reactor Coolant Flow	PHOENIX, LOFTRAN THINC, FACTRAN	---	0	upper	2396 and 3423
Waste Gas Decay Tank Rupture	NA	---	NA	NA	3577
Single RCC Assembly Withdrawal at Full Power	TURTLE, THINC LEOPARD	---	NA	NA	3423
CONDITION IV					
Major rupture of pipes containing reactor coolant up to and including double-ended rupture of the largest pipe in the Reactor Coolant System (Loss of Coolant Accident)	SATAN LOCTA-R2	Function of Moderator density See Section 15.4.1 Salem FSAR		Function of Fuel Temp. See Section 15.4.1 Salem FSAR	3579
Major secondary system pipe rupture up to and including double-ended rupture (Rupture of a Steam Pipe)	MARVEL, THINC	Function of Moderator Density See Section 15.2.13 (Fig. 15.2-41) Salem FSAR		-2.2 pcm/°F	0 (Subcritical)

TABLE 3.2 (Sheet 4 of 4)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

FAULTS	COMPUTER CODES USED	REACTIVITY COEFFICIENTS ASSUMED: MODERATOR[1] TEMPERATURE ($\Delta k/°F$)	MODERATOR[1] DENSITY ($\Delta K/gm-cc$)	DOPPLER[2]	INITIAL NSSS THERMAL POWER OUTPUT ASSUMED (MWt)
CONDITION IV (Cont'd.)					
Steam Generator Tube Rupture	NA	NA	NA	NA	3577
Single Reactor Coolant Pump Locked Rotor	PHOENIX, LOFTRAN THINC, FACTRAN	---	0	upper	2396 and 3423
Fuel Handling Accident	NA	NA	NA		3577
Rupture of a Control Rod Mechanism Housing (RCCA Ejection)	TWINKLE, FACTRAN LEOPARD	-1 pcm/°F BOL -26 pcm/°F EOL	---	Consistent with lower limit shown Fig. 15.1-5	0 and 3423

NOTES:

(1) Only one is used in an analysis i.e., either moderator temperature or moderator density coefficient

(2) Reference Figure 15.1-5 of Salem FSAR

4.0 EQUIPMENT REVIEW

4.1 DESIGN TRANSIENTS

Design transients originally used for Salem Unit 1 and 2 components are contained in Westinghouse System Standard Design Criteria 1.3 Rev. 1, Nuclear Steam Supply System Design Transients. These transients were conservatively based on a thermal power level in excess of the 3423 MWt rating and, therefore, remain bounding for the Salem 1 uprating. The above statement is valid based on the following:

1) The original licensing basis remains applicable;
2) The same assumptions remain applicable; and
3) The same calculational techniques are applicable.

4.2 PRIMARY PLANT COMPONENTS

4.2.1 REACTOR VESSEL

To assess the impact of the uprating on the reactor vessel design and operation, the vessel design specification, stress report, and fracture mechanics analyses were considered.

The reactor vessel design specification was reviewed and the changes associated with the increased temperatures and power rating were incorporated.

The reactor vessel stress report was reviewed and where appropriate updated to reflect the duty cycle associated with the uprating. In all instances the updated data for stress intensity range and usage factor remain in compliance with the original design bases limits.

A review of the reactor vessel fracture mechanics evaluation revealed that the currend end-of-life fluence calculation was based on a core power of 3483 MWt. Therefore, the uprated power level fluence is bounded by the current analysis and the currently docketed data remains applicable.

4.2.2 REACTOR VESSEL INTERNALS

Review of the Unit 1 reactor vessel lower internals and comparison with Unit 2 revealed a design difference in the baffle barrel region former flow holes. Unit 1 was designed with flow holes of varying diameter in each former and from one former level to the next. Unit 2 was designed with uniform diameter holes in each former and at each former level.

Because of this difference it was necessary that a thermal-hydraulic analysis of the internals be performed to evaluate the effects of the power uprating. The results of this hydraulic review indicated negligible differences in core bypass flow and pressure drops and therefore the current data remain bounding for operation of the Unit 1 internals at the Unit 2 power level. In addition, hydraulic lift forces and the RCCA scram time changed insignificantly.

A thermal stress analysis was performed for the lower internals to assess the impact of the higher core radiation heat generation rates associated with the uprating. The increased component gamma heating was accounted for by increasing the heat generation rates to the level corresponding to the uprated power level. The results indicate that the radial, axial and azimuthal temperatures change by less than 2°F in the core baffle plates and core barrel. This results in a negligible effect on the performance of the internals components.

Unit 1 and Unit 2 also have differing control rod guide tube arrangements which were determined to be inconsequential with respect operation at the uprated conditions.

Based upon the above results, the Salem Unit 1 uprating to 3423 MWt has been demonstrated to be acceptable and remain in compliance with the original design criteria and bases.

4.2.3 REACTOR COOLANT PUMPS AND CONTROL ROD DRIVE MECHANISMS

For the 3423 MWt uprating program, the cold leg temperature at 100 percent power increases from 544.4°F to 545.0°F and the hot leg temperature increases from 609.1°F to 610.8°F. The component design pressure and temperature conditions and design transients are unchanged. The proposed 3423 MWt operating temperatures are bounded by those contained in the original equipment specifications and associated stress reports, therefore, no additional thermal and structural analysis were required at 3423 MWt for the reactor coolant pumps and control rod drive mechanisms to demonstrate compliance with codes and standards applicable to Salem Unit 1. The 3423 MWt uprating changes are bounded by the existing thermal and structural analysis. No equipment modifications or revised operating limits associated with these components are appropriate or necessary at the uprated conditions.

4.2.4 REACTOR COOLANT PIPING

Review of the thermal analysis of the reactor coolant loop, piping and supports was conducted for all Class I piping systems in Westinghouse scope.

1. All piping stresses for the uprated conditions are enveloped by those documented in the current stress report. The original design contained adequate margin to envelope the uprated conditions;

2. All support loads are within engineering tolerances of the original loads and require no additional examination. No reanalysis was necessary for the implementation of the uprating.

4.2.5 PRESSURIZER

Each pressurizer component analyzed in the original stress report was reviewed to identify any modifications required for operation at uprated conditions. Results of the evaluation showed that the existing pressurizer stress report satisfies all applicable ASME Code requirements when the Unit 1 parameters are revised to reflect the enveloping plant operating parameters of the uprating. The impact of the uprating on the pressurizer is bounded, and no equipment modification was required.

A separate review was performed to assess the adequacy of the pressurizer spray, safety and power operated relief valves for operation at 3423 MWt. Results indicate that the presently installed valve capacities were based on the Unit 2, 3423 MWt, power level and are sufficient for the uprated conditions.

4.2.6 STEAM GENERATORS

Since both Salem Unit 1 and 2 have Model 51 Series S/G's, the uprating of Unit 1 to the power level of Unit 2 is enveloped by the current design bases and a new stress analysis was not required. Thus the Unit 1 steam generator design is adequate for the uprated power level.

4.3 AUXILIARY EQUIPMENT

The auxiliary equipment for Salem Unit 1 affected by the uprated parameters has been reviewed and found to be functionally or physically identical to Unit 2.

4.3.1 AUXILIARY VALVES

The Salem 1 safety, POR, pressurizer spray and auxiliary valves were evaluated for a power uprating to the Unit 2 power of 3423 MWt. Westinghouse records indicate the valves for both units are identical and hence the Unit 1 valves will operate acceptably at the Unit 2 parameters.

4.3.2 AUXILIARY PUMPS

Auxiliary pumps for Salem Unit 1 were designed and supplied to the same design and functional requirements as Salem Unit 2. Therefore, these pumps will support a power uprating equivalent to the Unit 2 rating of 3423 MWt without modification or alterations.

4.3.3 AUXILIARY HEAT EXCHANGERS

Auxiliary heat exchangers (e.g., regenerative, non-regenerative, seal water return, and RHR heat exchangers) for Salem Unit 1 were designed and supplied to the same design and functional requirements as Salem Unit 2. Therefore, these heat exchangers are adequate and capable of performing at the uprated power of 3423 MWt.

4.3.4 AUXILIARY TANKS, DEMINERALIZERS AND FILTERS

Auxiliary tanks, demineralizers and filters for Salem Unit 1 were designed and supplied to the same design and functional requirements as Salem Unit 2. Therefore this equipment is not significantly affected by the Unit 1 uprating.

5.0 BALANCE OF PLANT

5.1 INTRODUCTION

To coordinate the NSSS review with the Balance of Plant (BOP) reviews, data provided by Westinghouse as input to the BOP design was examined to identify those areas where revisions might be required. Westinghouse conducted a review to verify the bases and to confirm the continued applicability of data originally supplied to PSE&G considering the NSSS uprating to 3423 MWt. The following sections describe the more significant areas where design data was confirmed by Westinghouse.

5.2 MASS AND ENERGY RELEASE

The original loss of coolant accident mass and energy release data provided as input for containment integrity was based upon a power rating of 3579 MWt. This data bounds the power increase to 3423 MWt. The main steamline break data was based on the event occurring at the no load condition which is unchanged.

5.3 AUXILIARY FEEDWATER SYSTEM

The original auxiliary feedwater system is identical to the auxiliary feedwater system in Salem Unit 2 which is currently operating at 3423 MWt/805 psia steam. Since the auxiliary feedwater requirements presented in the Salem FSAR cover both Salem Units 1 and 2 and were designed to 3577 MWt, the auxiliary feedwater system requirements ~~review~~ remain unchanged.

5.4 SOURCE TERMS FOR OFFSITE DOSE EVALUATIONS

The data in the current Salem FSAR are based upon a core power of 3558 MWt. The source terms are essentially only a function of core power and burnup. The increase in core power from 3338 MWt to 3411 MWt is still within the bounds of the Salem FSAR. The current Salem FSAR data remain valid.

5.5 SPENT FUEL PIT DECAY HEAT LOADS

The decay heat of the fuel in the spent fuel pit is a function of core power and burnup. Each Salem Unit has completely independent spent fuel pit cooling systems. The components and systems are identical for both units. Since Salem Unit 2 is presently operating at 3423 MWt and the requirements presented in the Salem FSAR remain unchanged, the data employed in the original design remain unchanged.

5.6 STEAM SYSTEM DESIGN TRANSIENTS

The steam system transients provided in the Westinghouse Steam Systems Design Manual are unchanged.

5.7 RCS LOOP PIPE LOADS, THERMAL DISPLACEMENTS AND DESIGN DATA

Based on a detailed review, it was determined that any changes in loadings and piping/support thermal displacements and other design data are within the bounds of the original evaluation.

5.8 CONDENSATE AND FEEDWATER SYSTEMS

The condensate and feedwater systems in Salem Unit 1 are identical to the condensate and feedwater systems in Salem Unit 2 which is presently operating at 3423 MWt/805 psia steam. Since the requirements presented in the Salem FSAR cover both Salem Units 1 and 2, the condensate and feedwater systems requirements remain unchanged.

5.9 MAIN STEAM SYSTEM

The main steam system for Salem Unit 1 was designed at the higher Salem Unit 2 design flowrate of 3.7×10^6 pounds per hour for each steam generator. The current Salem FSAR data remain valid.

5.10 COMPONENT COOLING WATER SYSTEM

Salem Units 1 and 2 have similar component cooling systems. They differ only in that Salem Unit 1 has one tube and shell type heat exchanger and one plate type heat exchanger whereas Salem Unit 2 has two tube and shell type heat exchangers. This difference is inconsequential as each type has a design heat transfer rate of 44.2×10^6 BTU/hr. Salem Unit 2 is presently operating at 3423 MWt/805 psia steam. The code requirements and minimum flow requirements presented in the Salem FSAR are applicable to both units. Therefore the component cooling system for Salem Unit 1 is acceptable for operation at 3423 MWt/805 psia steam.

5.11 TURBINE

Based on a review, it was determined that the assumptions, analyses, and evaluations performed to verify the operating characteristics and structural integrity of the turbine bound operation at 3423 MWt/805 psia steam.

6.0 FLUID SYSTEMS REVIEW

6.1 RESULTS

The Salem Unit 1 uprating engineering implementation program included assessments of the following systems: the Reactor Coolant System, Safety Injection System, Chemical and Volume Control System, Residual Heat Removal System, Spent Fuel Pit Cooling System and Sampling System.

The systems design documentation was reviewed and found to be consistent with the uprated power level of Unit 2. The review confirmed that the fluid system design parameters of Unit 2 (at 3423 MWt NSSS power) were in fact used to size the systems and equipment for both units.

All of the above mentioned systems will perform their functions at the uprated power in accordance with all current Salem Unit 1 design basis and criteria.

7.0 NUCLEAR FUEL REVIEW

7.1 RESULTS

The review to determine the effects of the Salem Unit 1 uprating to 3411 MWt nuclear power on the fuel design was conducted on a basis consistent with PSE&G's desires to convert Unit 1 to 18 month fuel cycles. Both the fuel currently in the core and reload fuel were reviewed with respect to nuclear design, thermal hydraulic design and fuel performance.

The nuclear design of the core at 3411 MWt will be addressed as part of the normal design process for Cycle 7.

The thermal hydraulic design of Salem Unit 1 is already being performed at 3411 MWt core power. No additional effort was required to verify acceptability.

An analysis of the impact of the Salem Unit 1 uprating has indicated that sufficient margin exists in the design of the fuel presently in the core to permit operation at the uprated power level. Reload fuel design analyses have been amended using the 3411 MWt core power design requirements. The fuel presently in the core is capable of being operated at either 3411 MWt or at the current 3338 MWt core power level.

8.0 FSAR CHANGES

The following describes the required revisions to the FSAR to reflect the change in power rating of Salem Unit 1 from 3350 MWt NSSS power to 3423 MWt NSSS power. In accordance with PSE&G's request, handwritten mark-ups of the revised FSAR pages are included in appendices 8A and 8B.

8.1 FSAR

SECTION 4.1

1. Table 4.1-1A (Sheets 1, 2, and 3) should be replaced with the attached sheets.

2. Table 4.1-1A (Sheet 6 of 6) contains table headings with no information. This page should be eliminated and the footnotes moved to the previous pages of the Table.

SECTION 4.3

1. In Section 4.3.1.1, Fuel Burnup, under Basis, the average region discharge burnup of 33,000 MWD/MTU should be changed to 38,000 MWD/MTU to reflect the current contract discharge burnup. This revision is not a result of the uprating.

2. In Section 4.3.2.1, Nuclear Design Description, the average region discharge burnup of 33,000 MWD/MTU should be changed to 38,000 MWD/MTU to reflect the current contract discharge burnup. This revision is not a result of the uprating.

3. In Section 4.3.2.2.6, Limiting Power Distributions, the average kw/ft should be changed from 5.33 kw/ft to 5.44 kw/ft. This is the resultant change in linear power density in uprating from 3338 MWt to 3411 MWt.

SECTION 4.4

1. Tables 4.4-1A, 4.4-2A, and 4.4-3A should be replaced with the attached sheets.

8.2 TECHNICAL SPECIFICATIONS

1. In Section 1.0 Definitions, subsection 1.3 Rated Thermal Power, 3338 MWt should be changed to 3411 MWt which incorporates the uprated power level.

2. In Table 3.2-1, DNB parameters (pg. 3/4 2-14), change Reactor Coolant System Tavg to 582°F. Delete 3 Loops in operation limits.

APPENDIX 8A

FSAR PAGE REVISIONS
HAND MARKED

TABLE 4.1-1A (Sheet 1 of 6)

REACTOR DESIGN COMPARISON TABLE

Thermal and Hydraulic Design Parameters	Salem Unit 1 17x17 Fuel Assembly With Densification Effects	Salem Unit 1 15x15 Fuel Assembly Without Densification Effects
1. Reactor Core Heat Output, MWt	~~3338~~ 3411	~~3338~~ 3411
2. Reactor Core Heat Output, Btu/hr	11642 ~~11,393~~ x 10^6	11642 ~~11,393~~ x 10^6
3. Heat Generated in Fuel, %	97.4	97.4
4. System Pressure, Nominal, psia	2250	2250
5. System Pressure, Min. Steady State, psia	2220	2220
6. Minimum DNBR at Nominal Conditions Typical Flow Channel,	2.24 ~~2.31~~	2.00 ~~2.09~~[a]
Thimble (Cold Wall) Flow Channel	1.80 ~~1.86~~	---
7. Minimum DNBR for Design Transients	>1.30	>1.30
Coolant Flow		
8. Total Thermal Flow Rate, lb/hr	132.~~7~~2 x 10^6	134.~~1~~0 x 10^6
9. Effective Flow Rate for Heat Transfer, lb/hr	126.~~4~~3 x 10^6	128.0 x 10^6
10. Effective Flow Area for Heat Transfer, ft^2	51.1	51.2
11. Average Velocity Along Fuel Rods, ft/sec	15.3	15.5
12. Average Mass Velocity, lb/hr-ft^2	2.47 x 10^6	2.50 x 10^6

TABLE 4.1-1A (Sheet 2 of 6)

REACTOR DESIGN COMPARISON TABLE

Thermal and Hydraulic Design Parameters	Salem Unit 1 17x17 Fuel Assembly With Densification Effects	Salem Unit 1 15x15 Fuel Assembly Without Densification Effects
Coolant Temperature, °F		
13. Nominal Inlet	~~544.4~~ 545.0	~~544.4~~ 545.0
14. Average Rise in Vessel	~~64.7~~ 65.8	~~63.9~~ 65.1
15. Average Rise in Core	~~67.5~~ 68.7	~~66.6~~ 67.8
16. Average in Core	~~579.8~~ 581.0	~~579.1~~ 580.4
16. Average in Vessel	~~576.7~~ 577.9	~~576.3~~ 577.5
Heat Transfer		
18. Active Heat Transfer, Surface Area, ft^2	59,700	52,200
19. Average Heat Flux, $Btu/hr\text{-}ft^2$	~~185,700~~ 189700	~~212,600~~ 217200
20. Maximum Heat Flux for Normal Operation, $Btu/hr\text{-}ft^2$	440200 ~~430,900~~[b]	580,000
21. Average Thermal Output, kw/ft	~~5.33~~ 5.44	~~6.88~~ 7.03
22. Maximum Thermal Output for Normal Operation, kw/ft	~~12.4~~[b] 12.6	18.8
23. Peak linear power for determination of protection set-points, kw/ft.	18.0[d]	---
24. Heat Flux Hot Channel Factor F_Q	2.32[c]	2.40 ~~2.46~~

TABLE 4.1-1A (Sheet 3 of 6)

REACTOR DESIGN COMPARISON TABLE

Thermal and Hydraulic Design Parameters	Salem Unit 1 17x17 Fuel Assembly With Densification Effects	Salem Unit 1 15x15 Fuel Assembly Without Densification Effects
Fuel Central Temperature, °F		
25. Peak at 100 Percent Power	~~3350~~ 3400	4250
26. Peak at Maximum Thermal Output for Maximum Overpower Trip Point	4150	---
Core Mechanical Design Parameters		
Fuel Assemblies		
27. Design	RCC Canless	RCC Canless
28. Number of Fuel Assemblies	193	193
29. UO_2 Rods per Assembly	264	204
30. Rod Pitch, in.	0.496	0.563
31. Overall Dimensions, in.	8.426 x 8.426	8.426 x 8.426
32. Fuel Weight (as UO_2), pounds	222,739	215,400
33. Zircaloy Weight, lbs.	50,913	48,250
34. Number of Grids per Assembly	8-Type R	7-Type L
35. Loading Technique	3 region non-uniform	3 region non-uniform
Fuel Rods		
36. Number	50,952	39,372
37. Outside Diameter, in.	0.374	0.422
38. Diametral Gap, in., Regions 1, 2, (and 3)	0.0065	0.0075 (0.0085)

TABLE 4.1-1A (Sheet 6 of 6)

REACTOR DESIGN COMPARISON TABLE

Nuclear Design Parameters	Salem Unit 1 17x17 Fuel Assembly With Densification Effects	Salem Unit 1 15x15 Fuel Assembly Without Densification Effects

Move to previous Page

[a] Previously, the value of 2.09 for a limiting typical channel was quoted only since the thimble (cold wall) DNB tests were incomplete.
[b] This limit is associated with the value of F_Q = 2.32.
[c] Includes the effect of fuel densification.
[d] See Section 4.3.2.2.6.

Condition III incidents shall not cause more than a small fraction of the fuel elements in the reactor to be damaged, although sufficient fuel element damage might occur to preclude immediate resumption of operation. The release of radioactive material due to Condition III incidents should not be sufficient to interrupt or restrict public use of these areas beyond the exclusion radius. Furthermore, a Condition III incident shall not, by itself generate a Condition IV fault or result in a consequential loss of function of the Reactor Coolant System or reactor containment barriers.

Condition IV occurrences are faults that are not expected to occur but are defined as limiting faults which must be designed against. Condition IV faults shall not cause a release of radioactive material that results in an undue risk to public health and safety.

The core design power distribution limits related to fuel integrity are met for Condition I occurrences through conservative design and maintained by the action of the Control System. The requirements for Condition II occurrences are met by providing an adequate protection system which monitors reactor parameters. The Control and Protection Systems are described in Chapter 7, and the consequences of Condition II, III and IV occurrences are given in Chapter 15.

4.3.1.1 <u>Fuel Burnup</u>

Basis

The fuel rod design basis is described in Section 4.2. The nuclear design basis is to install sufficient reactivity in the fuel to attain an average region discharge burnup of ~~33,000~~ 38000 MWD/MTU. The above, along with the design basis in Section 4.3.1.3, Control of Power Distribution, satisfies GDC-10.

fuel on the core periphery, with depleted fuel moved inward. The cores will normally operate approximately 11,000 MWD/MTU per year. The enrichments for the first cores are shown in Table 4.3-1.

The core average enrichment is determined by the amount of fissionable material required to provide the desired core lifetime and energy requirements, namely a region average discharge burnup of 38000 ~~33,000~~ MTD/MTU. The physics of the burnout process is such that operation of the reactor depletes the amount of fuel available due to the absorption of neutrons by the U-235 atoms and their subsequent fission. The rate of U-235 depletion is directly proportional to the power level at which the reactor is operated. In addition, the fission process results in the formation of fisson products, some of which readily absorb neutrons. These effects, depletion and the buildup of fission products, are partially offset by the buildup of plutonium shown in Figure 4.3-2 for the 17 x 17 fuel assembly, which occurs due to the non-fission absorption of neutrons in U-238. Therefore, at the beginning of any cycle a reactivity reserve equal to the depletion of the fissionable fuel and the buildup of fission product poisons over the specified cycle life must be "built" into the reactor. This excess reactivity is controlled by removable neutron absorbing material in the form of boron dissolved in the primary coolant and burnable poison rods.

The concentration of boric acid in the primary coolant is varied to provide control and to compensate for long-term reactivity requirements. The concentration of the soluble neutron absorber is varied to compensate for reactivity changes due to fuel burnup, fission product poisoning including xenon and samarium, burnable poison depletion, and the cold-to-operating moderator temperature change. Using its normal makeup path, the Chemical and Volume Control System (CVCS) is capable of inserting negative reactivity at a rate of approximately 30 pcm/min when the reactor coolant boron concentration is 1000 ppm and approximately 35 pcm/min when the reactor coolant boron concentration is 100 ppm. The peak burnout rate for xenon is 25 pcm/min. Rapid transient reactivity

Allowing for fuel densification effects the average kw/ft for Unit No. 1 and Unit No. 2 is 5.44 kw/ft. ~~is 5.33 kw/ft and 5.44 for Unit No. 2~~. From Figure 4.3-20, the conservative upper bound value of normalized local power density, including allowances for densification effects, is 2.32 corresponding to a peak local power density of ~~12.6 kw/ft and~~ 12.9 kw/ft at 102 percent power for Unit No. 1 and Unit No. 2. ~~respectively.~~

To determine Reactor Protection System set points, with respect to power distributions, three categories of events are considered, namely rod control equipment malfunctions, operator errors of commission and operator errors of omission.

The first category comprises uncontrolled rod withdrawal (with rods moving in the normal bank sequence) for full length rod banks. Also included are motions of the full length rod banks below their insertion limits, which could be caused, for example, by uncontrolled dilution or primary coolant cooldown. Power distributions were calculated throughout these occurrences assuming short term corrective action, that is no transient xenon effects were considered to result from the malfunction. The event was assumed to occur from typical normal operating situations which did include normal xenon transients. It was further assumed in determining the power distributions that total power level would be limited by reactor trip to below 118 percent. Since the study is to determine protection limits with respect to power and axial offset, no credit was taken for trip set point reduction due to flux difference. Results are given in Figure 4.3-21 in units of kw/ft. The peak power density which can occur in such events, assuming reactor trip at or below 118 percent, is thus limited to 18.0 kw/ft including uncertainties and densification effects. The second category, also appearing in Figure 4.3-21, assumes that the operator mis-positions the full length rod bank in violation of the insertion limits and creates short term conditions not included in normal operating conditions.

TABLE 4.4-1A (sheet 1 of 2)

REACTOR DESIGN COMPARISON TABLE SALEM UNIT 1

Thermal and Hydraulic Design Parameters	17 x 17 With Densification	15 x 15 Without Densification
Reactor Core Heat Output, MWt	~~3338~~ 3411	~~3338~~ 3411
Reactor Core Heat Output, Btu/hr	11642 ~~11,393~~ x 10^6	11642 ~~11,393~~ x 10^6
Heat Generated in Fuel,	97.4	97.4
System Pressure, Nominal, psia	2250	2250
System Pressure, Minimum Steady State, psia	2220	2220
Minimum DNBR at Nominal Conditions		
Typical Flow Channel	~~2.31~~ 2.24	2.00 ~~2.09~~[a]
Thimble (Cold Wall) Flow Channel	~~1.86~~ 1.80	---
Minimum DNBR for Design Transients	>1.30	>1.30
DNB Correlation	"R-Grid" (W-3 with modified spacer factor)	"R-Grid" (W-3 with modified spacer factor)
Coolant Flow		
Total Thermal Flow Rate, lb/hr	132.~~3~~2 x 10^6	134.~~1~~0 x 10^6
Effective Flow Rate for Heat Transfer, lb/hr	126.~~4~~3 x 10^6	128.0 x 10^6
Effective Flow Area for Heat Transfer, ft^2	51.1	51.2
Average Velocity Along Fuel Rods, ft/sec	15.~~3~~4	15.~~8~~6
Average Mass Velocity, lb/hr-ft^2	2.47 x 10^6	2.50 x 10^6
Coolant Temperature		
Nominal Inlet, °F	~~544.4~~ 545.0	~~544.4~~ 545.0
Average Rise in Vessel, °F	~~64.7~~ 65.8	~~63.9~~ 65.1
Average Rise in Core, °F	~~67.5~~ 68.7	~~66.6~~ 67.8

TABLE 4.4-1A (sheet 2 of 2)

REACTOR DESIGN COMPARISON TABLE SALEM UNIT 1

Thermal and Hydraulic Design Parameters	17 x 17 With Densification	15 x 15 Without Densification
Average in Core, °F	~~579.8~~ 581.0	~~579.1~~ 580.4
Average in Vessel, °F	~~576.7~~ 577.9	~~576.3~~ 577.5
Heat Transfer		
Active Heat Transfer, Surface Area, ft^2	59,700	52,200
Average Heat Flux, Btu/hr-ft^2	189700 ~~185,700~~	217200 ~~212,600~~
Maximum Heat Flux, for normal operation Btu/hr-ft^2	440200 ~~430,900~~[b]	580,000
Average Thermal Output, kw/ft	5.44 ~~5.33~~	7.03 ~~6.88~~
Maximum Thermal Output, for normal operation, kw/ft	12.6 ~~12.4~~[b]	18.8
Peak Linear Power for determination of Protection Setpoints, kw/ft	18.0[c]	---
Fuel Central Temperature		
Peak at 100 Power, °F	3400 ~~3350~~	4250
Peak at Thermal Output Maximum for Maximum Overpower Trip Point, °F	4150	
Pressure Drop		
Across Core, psi	24.7 ± 2.5[d]	32.6[e]
Across Vessel, including nozzle, psi	49.8 ± 5.0	52.0

[a] Previously, the value of 2.09 for a limiting typical channel was quoted only since the thimble (cold wall) DNB Tests were incomplete.
[b] This limit is associated with the value of F_Q = 2.32.
[c] See Section 4.3.2.2.6.
[d] Based on best estimate reactor flow rate of 95,600 gpm/loop.
[e] Previously, a conservatively high value of pressure drop was used to determine vessel loop flow rates.

TABLE 4.4-2A

THERMAL-HYDRAULIC DESIGN PARAMETERS FOR
ONE OF FOUR COOLANT LOOPS OUT OF SERVICE
SALEM UNIT 1

Total Core Heat Output, MWt	~~2337~~ 2388
Total Core Heat Output, 10^6 Btu/hr	~~7976~~ 8150
Heat Generated in Fuel,	97.4
Nominal System Pressure, psia	2250
Coolant Flow	
Effective Thermal Flow Rate for Heat Transfer, 10^6 lbs/hr	90.6
Effective Flow Area for Heat Transfer, ft^2	51.1
Average Velocity along Fuel Rods, ft/sec	10.9
Average Mass Velocity, 10^6 $lb/hr\text{-}ft^2$	1.77
Coolant Temperature, °F	
Design Nominal Inlet	~~538.8~~ 539.1
Average Rise in Core	~~67.0~~ 68.2
Average in Core	~~573.8~~ 574.7
Heat Transfer	
Active Heat Transfer Surface Area, ft^2	59,700
Average Heat Flux, $Btu/hr\text{-}ft^2$	130,000
Minimum DNB Ratio at Nominal Conditions	> ~~1.86~~ 1.80
Minimum DNB Ratio for Design and Anticipated Transients	$\geq$ 1.30

TABLE 4.4-3A

VOID FRACTIONS AT NOMINAL REACTOR CONDITIONS

WITH DESIGN HOT CHANNEL FACTORS

SALEM UNIT 1

	Average	Maximum
Core	0.18 ~~0.15~~%	--
Hot Subchannel	4.0 ~~0.9~~%	13.6 ~~2.2~~