APPENDIX 8B

TECHNICAL SPECIFICATION REVISIONS

HAND MARKED

1.0 DEFINITIONS

DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

1.2 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of ~~2338~~ MWt.
3411

OPERATIONAL MODE

1.4 An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principle specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxil equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their rela support function(s).

REPORTABLE OCCURRENCE

1.7 A REPORTABLE OCCURRENCE shall be any of those conditions specified in Specifications 6.9.1.8 and 6.9.1.9.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS: 4 Loops In Operation	LIMITS: ~~3 Loops in Operation~~
Reactor Coolant System T_{avg}	≤ ~~581~~ 582°F	~~≤ 572°F~~
Pressurizer Pressure	≥ 2220 psia*	~~≥ 2220 psia*~~
~~Reactor Coolant System~~	~~≥ 349,200 gpm~~	~~≥ 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.

ATTACHMENT # 2

ATTACHMENT # 2

DRAFT FSAR CHANGES TO BE INCORPORATED UPON
ISSUANCE OF THE SALEM UNIT 1 POWER UPRATE AMENDMENT

The following changes have been identified:

Chapter 1 - Change page 1.0-1 as per the attached marked-up page.

Chapter 2&3 - No Changes

Chapter 4* - Delete Tables 4.1-1A (six pages), 4.4-1A (two pages) 4.4-2A (one page, and 4.4-3A (one page).

- Change Tables 4.1-1B (six pages), 4.4-1B (two pages), 4.4-2B (one page), and 4.4-3B (one page) as per the attached marked-up pages.

- Change throughout Section 4.4 any reference to "Tables 4.4-1A and B" to "Table 4.4-1". Similarly for Tables 4.4-2A and B, and Tables 4.4-3A and B.

- Change pages 4.4-5, 4.4-13, and 4.4-54 as per the attached marked-up pages.

pter 5 - Revise Tables 5.1-1 (one page), 5.2-3 (two pages), 5.2-5 (two pages), 5.2-7 (one page) and 5.5-1 (one page) per the attached marked-up pages.

Chapters 6, 7, 8, and 9 - No Changes.

Chapter 10 - Revise pages 10.2-1 and 10.3-2 per the attached marked up pages.

Chapters 11, 12, 13, 14 and 15 - No Changes.

*Section 4.3, Nuclear Design, is based on Cycle 1 core design. Since the uprate will not be implemented until cycle 7, it is not appropriate to change Section 4.3 to reflect the uprated conditions. Revisions to the Nuclear Design are addressed by the reload analyses, Section 4.5.

1.0 - INTRODUCTION AND SUMMARY

This Updated Final Safety Analysis Report is submitted pursuant to the requirements of 10 CFR 50.71 by Public Service Electric and Gas Company (PSE&G) for the two nuclear power units at its Salem Generating Station.

PSE&G and Westinghouse Electric Corporation have jointly participated in the design and construction of each unit. The plant is operated by PSE&G. Each unit employs a pressurized water reactor nuclear steam supply system furnished by Westinghouse which is similar in design concept to several other projects licensed by the Nuclear Regulatory Commision. The only systems shared by the two units are Compressed Air, Demineralized Water and the Solid Radwaste Handling System. There are a minimum of shared components; chemical drain and laundry hot shower tanks and pumps are the only components in common.

The licensed core power for both units is 3411 MWt. ~~ratings of the two units are as follows: Unit 1 3338 MWt, and Unit 2 - 3411 MWt~~. The ~~warranted~~ gross and approximate net electrical outputs are ~~1132 MWe and 1090 MWe respectively for Unit 1 and~~ 1158 MWe and 1115 MWe respectively ~~for Unit 2~~. The reactors are expected to be capable of outputs of approximately ~~3494 MWt (Unit 1) and~~ 3570 MWt ~~(Unit 2)~~, which corresponds to the valves-wide-open rating of the turbine generators of ~~1176 MWe gross and 1130 MWe net for Unit 1, and~~ 1201 MWe gross and 1155 MWe net* ~~for Unit 2~~. The containment and engineered safety features for both units have been designed and evaluated at the ~~Unit 2~~ maximum power rating of 3570 Mwt. Most postulated accidents have been evaluated at 3423 MWt. Loss-of-coolant accidents and those postulated accidents having offsite dose consequences have been analyzed at the power rating of 3570 Mwt.

* Unit 1 turbine-generator maximum calculated load is presently 1178 MWe gross.

DELETE

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	3338
2. Reactor Core Heat Output, Btu/hr	$11,393 \times 10^6$	$11,393 \times 10^6$
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,	2.31	2.09[a]
Thimble (Cold Wall) Flow Channel	1.86	---
7. Minimum DNBR for Design Transients	>1.30	>1.30
Coolant Flow		
8. Total Thermal Flow Rate, lb/hr	132.3×10^6	134.1×10^6
9. Effective Flow Rate for Heat Transfer, lb/hr	126.4×10^6	128.0×10^6
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	2.47×10^6	2.50×10^6

DELETE

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	544.4
14. Average Rise in Vessel	64.7	63.9
15. Average Rise in Core	67.5	66.6
16. Average in Core	579.8	579.1
16. Average in Vessel	576.7	576.3
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	212,600
20. Maximum Heat Flux for Normal Operation, $Btu/hr\text{-}ft^2$	430,900[b]	580,000
21. Average Thermal Output, kw/ft	5.33	6.88
22. Maximum Thermal Output for Normal Operation, kw/ft	12.4[b]	18.8
23. Peak linear power for determination of protection setpoints, kw/ft.	18.0[d]	---
24. Heat Flux Hot Channel Factor F_Q	2.32[c]	2.16

DELETE

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	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)

DELETE

TABLE 4.1-1A (Sheet 4 of 6)

REACTOR DESIGN COMPARISON TABLE

Core Mechanical Design Parameters	Salem Unit 1 17x17 Fuel Assembly With Densification Effects	Salem Unit 1 15x15 Fuel Assembly Without Densification Effects
Fuel Rods (Cont'd)		
39. Clad Thickness, in.	0.0225	0.024
40. Clad Material	Zircaloy-4	Zircaloy-4
Fuel Pellets		
41. Material	UO_2 Sintered	UO_2 Sintered
42. Density (% of Theoretical)	95	94
43. Diameter, in., Regions 1, 2, (and 3)	0.3225	0.3659 (0.3649)
44. Length, in.	0.530	0.600
Rod Cluster Control Assemblies		
45. Neutron Absorber	Ag-In-Cd	Ag-In-Cd
46. Clad Material	Type 304 SS-Cold Worked	Type 304 SS-Cold Worked
47. Clad Thickness, in.	0.0185	0.019
48. Number of Clusters	53	53
49. Number of Absorber Rods per Cluster	24	20

TABLE 4.1-1A (Sheet 5 of 6)

REACTOR DESIGN COMPARISON TABLE

Core Mechanical Design Parameters	Salem Unit 1 17x17 Fuel Assembly With Densification Effects	Salem Unit 1 15x15 Fuel Assembly Without Densification Effects
Core Structure		
50. Core Barrel, I.D./O.D., in.	148.0/152.5	148.0/152.5
51. Thermal Shield I.D./O.D., in.	158.5/164.0	158.5/164.0
Nuclear Design Parameters		
Structure Characteristics		
52. Core Diameter, in. (Equivalent)	132.7	132.7
53. Core Average Active Fuel Height, in.	143.7	144
Reflector Thickness and Composition		
54. Top - Water plus Steel, in.	~10	~10
55. Bottom - Water plus Steel, in.	~10	~10
56. Side - Water plus Steel, in.	~15	~15
57. H_2O/U, Molecular Ratio, Lattice (cold)	2.41	2.52
Feed Enrichment, w/o		
58. Region 1	2.25	2.25
59. Region 2	2.80	2.80
60. Region 3	[illegible]	[illegible]

DELETE

DELETE

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

[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.

TABLE 4.1-1 (Sheet 1 of 6)

REACTOR DESIGN COMPARISON TABLE

Thermal and Hydraulic Design Parameters	~~Salem Unit 2~~ 17x17 Fuel Assembly With Densification Effects	~~Salem Unit 2~~ 15x15 Fuel Assembly Without Densification Effects
1. Reactor Core Heat Output, MWt	3411	3411
2. Reactor Core Heat Output, Btu/hr	$11,642 \times 10^6$	$11,642 \times 10^6$
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 Channels,	2.24	2.0[a]
Thimble (Cold Wall) Flow Channel.	1.80	---
7. Minimum DNBR for Design Transients	$\geq$ 1.30	$\geq$ 1.30
Coolant Flow		
8. Total Thermal Flow Rate, lb/hr	132.2×10^6	134.0×10^6
9. Effective Flow Rate for Heat Transfer, lb/hr	126.3×10^6	128.0×10^6
10. Effective Flow Area for Heat Transfer, ft^2	51.1	51.2
11. Average Velocity Along Fuel Rods, ft/sec	15.4	15.6
12. Average Mass Velocity, $lb/hr\text{-}ft^2$	2.47×10^6	2.50×10^6

TABLE 4.1-1X (Sheet 2 of 6)

REACTOR DESIGN COMPARISON TABLE

Thermal and Hydraulic Design Parameters	~~Salem Unit 2~~ 17x17 Fuel Assembly With Densification Effects	~~Salem Unit 2~~ 15x15 Fuel Assembly Without Densification Effects
Coolant Temperature, °F		
13. Nominal Inlet	545.0	545.0
14. Average Rise in Vessel	65.8	65.1
15. Average Rise in Core	68.7	67.8
16. Average in Core	581.0	580.4
17. Average in Vessel	577.9	577.5
Heat Transfer		
18. Active Heat Transfer, Surface Area, ft^2	59,700	52,200
19. Average Heat Flux, Btu/hr-ft^2	189,700	217,200
20. Maximum Heat Flux for Normal Operation, Btu/hr-ft^2	440,200[b]	580,000
21. Average Thermal Output, kw/ft	5.44	7.03
22. Maximum Thermal Output for Normal Operation, kw/ft	12.6[b]	18.8
23. Peak linear power for determination of protection set-points, kw/ft	18.0[c]	---

TABLE 4.1-1 (Sheet 3 of 6)

REACTOR DESIGN COMPARISON TABLE

Thermal and Hydraulic Design Parameters	~~Salem Unit 2~~ 17x17 Fuel Assembly With Densification Effects	~~Salem Unit 2~~ 15x15 Fuel Assembly Without Densification Effects
Heat Transfer (Cont'd)		
24. Heat Flux Hot Channel Factor, F_Q	2.32[c]	2.40
Fuel Central Temperature, °F		
25. Peak at 100 Percent Power	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 Dimension, 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

TABLE 4.1-1 (Sheet 4 of 6)

REACTOR DESIGN COMPARISON TABLE

Core Mechanical Design Parameters	~~Salem Unit 2~~ 17x17 Fuel Assembly With Densification Effects	~~Salem Unit 2~~ 15x15 Fuel Assembly Without Densification Effects
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)
39. Clad Thickness, in.	0.0225	0.0243
40. Clad Material	Zircaloy-4	Zircaloy-4
Fuel Pellets		
41. Material	UO_2 Sintered	UO_2 Sintered
42. Density (% of Theoretical)	95	94, 93, 92
43. Diameter, in., Regions 1, 2, (and 3)	0.3225	0.3659 (0.3649)
44. Length, in.	0.530	0.600
Rod Cluster Control Assemblies		
45. Neutron Absorber	Ag-In-Cd	Ag-In-Cd
46. Cladding Material	Type 304 SS-Cold Worked	Type 304 SS-Cold Worked
47. Clad Thickness, in.	0.0185	0.019
48. Number of Clusters	53	53

TABLE 4.1-1X (5 of 6)

REACTOR DESIGN COMPARISON TABLE

Core Mechanical Design Parameters	~~Salem Unit 2~~ 17x17 Fuel Assembly With Densification Effects	~~Salem Unit 2~~ 15x15 Fuel Assembly Without Densification Effects
Rod Cluster Control Assemblies (Cont'd)		
49. Number of Absorber Rods per Cluster	24	20
Core Structure		
50. Core Barrel, I.D./O.D., in.	148.0/152.5	148.0/152.5
51. Thermal Shield, I.D./O.D., in.	158.5/164.0	158.5/164.0
Nuclear Design Parameters		
Structure Characteristics		
52. Core Diameter, in. (Equivalent)	132.7	132.7
53. Core Average Active Fuel Height, in.	143.7	144
Reflector Thickness and Composition		
54. Top - Water plus Steel, in.	~10	~10
55. Bottom - Water plus Steel, in.	~10	~10
56. Side - Water plus Steel, in.	~15	~15
57. H_2O/U, Molecular Ratio, Lattice (cold)	2.41	2.52

TABLE 4.1-1~~X~~ (6 of 6)

REACTOR DESIGN COMPARISON TABLE

Nuclear Design Parameters	UNIT 1/UNIT 2 ~~Salem Unit 2~~ 17x17 Fuel Assembly With Densification Effects	UNITS 1 & 2 ~~Salem Unit 2~~ 15x15 Fuel Assembly Without Densification Effects
Feed Enrichment, w/o [e]		
58. Region 1	2.25 / 2.10	2.25
59. Region 2	2.80/ 2.60	2.80
60. Region 3	3.30/ 3.10	3.30

[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.
[e] Cycle 1 fuel

4.4.2.1 Summary Comparison

The design of the Salem Unit 1 and Unit 2 reactors with the 17 x 17 fuel rod array per assembly has the following identical thermal and hydraulic parameters as the 15 x 15 fuel rod array reactor design.

1. Core power

2. System pressure

3. Coolant inlet temperature

4. Open lattice fuel rod array

The vessel loop flow rates for both Unit 1 and Unit 2 thermal design are approximately 1.4 percent less than the 15 x 15 design valves. The basis for this change is discussed in Chapter 5. This change in flow also results in small changes in the core and vessel coolant average temperature and core and vessel coolant exit temperatures.

Values of each parameter are presented in Table 4.4-1 ~~Tables 4.4-1A* and B~~ for all coolant loops in service and in Table 4.4-2 ~~Tables 4.4-2A and B~~ for all but one coolant loop in service. It is also noted that in this power capability evaluation, there has not been any change in the design criteria. The reactor is still designed to a minimum DNBR $\geq$ 1.30 as well as no fuel centerline melting during normal operation, operational transients and faults of moderate frequency.

typical of several places

~~* Where applicable the Figures and Tables in this section consist of two parts labeled "A" and "B", which refer to Units 1 and 2 respectively.~~

adequate heat transfer is provided between the fuel clad and the reactor coolant so that the core thermal output is not limited by considerations of the clad temperature. Figure 4.4-4 shows the axial variation of average clad temperature for the average power rod both at beginning and end-of-life.

Treatment of Peaking Factors

The total heat flux hot channel factor, F_Q, is defined by the ratio of the maximum to core average heat flux. As presented in Table 4.3-2 and discussed in Section 4.3.2.2.1, the design value F_Q for normal operation is 2.32, including fuel densification effects.

This results in peak local power of ~~12.4 kw/ft and~~ 12.6 kw/ft ~~for Units 1 and 2 respectively,~~ at full power conditions. As described in Section 4.3.2.2.6 the peak local power at the maximum overpower trip point is 18.0 kw/ft. The centerline temperature at this kw/ft must be below the UO_2 melt temperature over the lifetime of the rod, including allowances for uncertainties. The melt temperature of unirradiated UO_2 is 5080°F[1] and decreases by 58°F per 10,000 MWD/MTU. From Figure 4.4-2, it is evident that the centerline temperatures at the maximum overpower trip points for both units are far below those required to produce melting. Fuel centerline and average temperatures at rated (100 percent) power and at the maximum overpower trip point are presented in Table 4.4-1. ~~Tables 4.4-1A and B.~~ Typical

4.4.2.3 Critical Heat Flux Ratio or Departure from Nucleate Boiling Ratio and Mixing Technology

The minimum DNBR's for the rated power, design overpower and anticipated transient conditions are given in Table 4.4-1 ~~Tables 4.4-1A and 1B~~. The core average DNBR is not a safety related item as it is not directly related to the minimum DNBR in the core, which occurs at some elevation in the limiting flow channel. Similarly, the DNBR at the hot spot is not Typical main parameter which affects the DNBR. If the Salem Units 1 and 2 were operating at full power and nominal steady state conditions as specified in Table 4.4-1 ~~Tables 4.4-1A and B~~, a reduction in local mass velocity of ~~72 percent and~~ 69 percent ~~respectively~~ would be required to reduce the DNBR from ~~1.66 and~~ 1.80 to 1.30. The above mass velocity effect on the DNB correlation was based on the assumption of fully developed flow along the full channel length. In reality a local flow blockage is expected to promote turbulence and thus would likely not effect DNBR at all.

Coolant flow blockages induce local crossflows as well as promote turbulence. Fuel rod behavior is changed under the influence of a sufficiently high crossflow component. Fuel rod vibration could occur, caused by this crossflow component, through vortex shedding or turbulent mechanisms. If the crossflow velocity exceeds the limit established for fluidelastic stability, large amplitude whirling results. The limits for a controlled vibration mechanism are established from studies of vortex shedding and turbulent pressure fluctuations. Crossflow velocity above the established limits can lead to mechanical wear of the fuel rods at the grid support locations. Fuel rod wear due to flow induced vibration is considered in the fuel rod fretting evaluation (Section 4.2).

4.4.4 TESTING AND VERIFICATION

4.4.4.1 Tests Prior to Initial Criticality

A reactor coolant flow test, as noted in Item 5 of Table 13.3-1, is performed following fuel loading but prior to initial criticality. Coolant loop pressure drop data is obtained in this test. This data in conjunction with coolant pump performance information allows determination of the coolant flow rates at reactor operating conditions. This test verifies that proper coolant flow rates have been used in the core thermal and hydraulic analysis.

TABLE 4.4-1A (sheet 1 of 2)

delete

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	3338
Reactor Core Heat Output, Btu/hr	11,393 x 10^6	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	2.09[a]
Thimble (Cold Wall) Flow Channel	1.86	---
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 x 10^6	134.1 x 10^6
Effective Flow Rate for Heat Transfer, lb/hr	126.4 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	15.5
Average Mass Velocity, lb/hr-ft^2	2.47 x 10^6	2.50 x 10^6
Coolant Temperature		
Nominal Inlet, °F	544.4	544.4
Average Rise in Vessel, °F	64.7	63.9
Average Rise in Core, °F	67.5	66.6

delete

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	579.1
Average in Vessel, °F	576.7	575.8
Heat Transfer		
Active Heat Transfer, Surface Area, ft^2	59,700	52,200
Average Heat Flux, $Btu/hr\text{-}ft^2$	185,700	232,600
Maximum Heat Flux, for normal operation $Btu/hr\text{-}ft^2$	430,900[a]	530,000
Average Thermal Output, kw/ft	5.33	6.33
Maximum Thermal Output, for normal operation, kw/ft	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	3350	4250
Peak at Thermal Output Maximum for Maximum Overpower Trip Point, °F	4150	
Pressure Drop		
Across Core, psi	24.7 ± 2.5[d]	32.5[e]
Across Vessel, including nozzle, psi	49.3 ± 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 4.4-1X (sheet 1 of 2)

REACTOR DESIGN COMPARISON TABLE ~~SALEM UNIT 2~~

Thermal and Hydraulic Design Parameters	17 x 17 With Densification	15 x 15 Without Densification
Reactor Core Heat Output, MWt	3411	3411
Reactor Core Heat Output, Btu/hr	11,642 x 10^6	11,642 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	2.24	2.0[a]
Thimble (Cold Wall) Flow Channel	1.80	---
Minimum DNBR for Design Transients	$\geq$1.30	$\geq$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	132.2 x 10^6	134.0 x 10^6
Effective Flow Rate for Heat Transfer, lb/hr	126.3 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	15.4	15.6
Average Mass Velocity, lb/hr-ft^2	2.47 x 10^6	2.50 x 10^6
Coolant Temperature		
Nominal Inlet, °F	545.0	545.0
Average Rise in Vessel, °F	65.8	65.1
Average Rise in [illegible], °F	[illegible]	[illegible]

TABLE 4.4-1X (sheet 2 of 2)

REACTOR DESIGN COMPARISON TABLE ~~SALEM UNIT 2~~

Thermal and Hydraulic Design Parameters	17 x 17 With Densification	15 x 15 Without Densification
Average in Core, °F	581.0	580.4
Average in Vessel, °F	577.9	577.5
Heat Transfer		
Active Heat Transfer, Surface Area, ft^2	59,700	52,200
Average Heat Flux, $Btu/hr\text{-}ft^2$	189,700	217,200
Maximum Heat Flux, for normal operation, $Btu/hr\text{-}ft^2$	440,200[b]	580,000
Average Thermal Output, kw/ft	5.44	7.03
Maximum Thermal Output, for normal operation kw/ft	12.6[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	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	49.3 ± 5.0	52.0

[a] [illegible]. 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.

delete

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
Total Core Heat Output, 10^6 Btu/hr	7976
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
Average Rise in Core	67.0
Average in Core	573.8
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
Minimum DNB Ratio for Design and Anticipated Transients	$\geq$ 1.30

delete

TABLE 4.4-3A

VOID FRACTIONS AT NOMINAL REACTOR CONDITIONS
WITH DESIGN HOT CHANNEL FACTORS
SALEM UNIT 1

	Average	Maximum
Core	0.15%	--
Hot Subchannel	0.9%	2.2

TABLE 4.4-2X

THERMAL-HYDRAULIC DESIGN PARAMETERS FOR
ONE OF FOUR COOLANT LOOPS OUT OF SERVICE
~~SALEM UNIT 2~~

Total Core Heat Output, MWt	2388
Total Core Heat Output, 10^6 Btu/hr	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-ft^2	1.77
Coolant Temperature, °F	
Design Nominal Inlet	539.1
Average Rise in Core	68.2
Average in Core	574.7
Heat Transfer	
Active Heat Transfer Surface Area, ft^2	59,700
Average Heat Flux, Btu/hr-ft^2	132,900
Minimum DNB Ratio at Nominal Conditions	> 1.80
Minimum DNB Ratio for Design and Anticipated Transients	$\geq$ 1.30

TABLE 4.4-3

VOID FRACTIONS AT NOMINAL REACTOR CONDITIONS

WITH DESIGN HOT CHANNEL FACTORS

~~SALEM UNIT 2~~

	Average	Maximum
Core	0.18%	--
Hot Subchannel	4.0%	13.6

TABLE 5.1-1

SYSTEM DESIGN AND OPERATING PARAMETERS

	~~Unit 1~~	~~Unit 2~~
Plant design life, years	~~40~~	40
Number of heat transfer loops	~~4~~	4
Design pressure, psig	~~2485~~	2485
Nominal operating pressure, psig	~~2235~~	2235
Total system volume including pressurizer and surge line (ambient conditions), ft^3	~~12,612~~	12,612
System liquid volume, including pressurizer and surge line (ambient conditions), ft^3	~~11,892~~	11,892
Total heat output (100 percent power), Btu/hr	~~$11,431 \times 10^6$~~	$11,680 \times 10^6$
Reactor vessel coolant temperature at full power:		
Inlet, nominal, °F	~~544.4~~	545.0
Outlet, °F	~~608.3~~	610.2
Coolant temperature rise in vessel at full power, avg, °F	~~63.9~~	65.2
Total coolant flow rate, lb/hr	~~134.1×10^6~~	132.2×10^6 ~~133.9×10^6~~
Steam pressure at full power, psia	~~805~~	805

DELETE THIS NOTE WHEN DOING ACTUAL FSAR REVISION

Editorial note:

The revised coolant flow, 132.2 x 10^6 lb/hr, is the total coolant flow with the 17x17 fuel assemblies. The originally reported flow in Tables 5.1-1, 5.2-3, and 5.2-5 for the RCS were apparently based on the 15x15 fuel assembly design. Refer to Table 4.1-1 for comparison of old and new values, 15x15 fuel and 17x17 fuel, respectively.

TABLE 5.2-3 (Sheet 1 of 2)

REACTOR VESSEL DESIGN DATA

	~~Unit 1~~	~~Unit 2~~
Design/Operating Pressure, psig	~~2485/2235~~	2485/2235
Hydrostatic Test Pressure, psig	~~3107~~	3107
Design Temperature, °F	~~650~~	650
Overall Height of Vessel and Closure Heat, ft-in. (bottom head OD to top of control rod mechanism adapter	~~43-10~~	43-10
Thickness of Insulation, min., in.	~~3~~	3
Number of Reactor Closure Head Studs	~~54~~	54
Diameter of Reactor Closure Head Studs, in.	~~7~~	7
ID of Flange, in.	~~172.5~~	172.5
OD of Flange, in.	~~205~~	205
ID at Shell, in.	~~173~~	173
Inlet Nozzle ID, in	~~27-1/2~~	27-1/2
Outlet Nozzle ID, in.	~~29~~	29
Clad Thickness, min., in.	~~5/32~~	5/32
Lower Head Thickness, min., in. (base metal)	~~5-3/8~~	5-3/8
Vessel Belt-Line Thickness, min., in. (base metal)	~~8.5~~	8.5
Closure Heat Thickness, in.	~~7~~	7
Reactor Coolant Inlet Temperature, °F	~~544.4~~	545.0
Reactor Coolant Outlet Temperature, °F	~~608.3~~	610.2
Reactor Coolant Flow, lb/hr	~~134.1×10^6~~	132.2×10^6 ~~133.9×10^6~~
Total Water Volume Below Core, ft^3	~~1050~~	1050
Water Volume in Active Core Region, ft^3	~~665~~	665

TABLE 5.2-3 (Sheet 2 of 2)

REACTOR VESSEL DESIGN DATA

	~~Unit 1~~	~~Unit 2~~
Total Water Volume to Top of Core, ft^3	~~2164~~	2164
Total Water Volume to Coolant Piping Nozzles Centerline, ft^3	~~2929~~	2959
Total Reactor Vessel Water Volume, (with core and internals in place), ft^3	~~4945~~	4945
Total Reactor Coolant System Volume, ft^3	~~12,612~~	12,612

DELETE THIS NOTE
WHEN DOING ACTUAL FSAR changes

Editorial Comment: The volume to the nozzles centerline, 2929 ft^3, originally reported for Unit 1 was a typographical error. The correct value is 2959, the same as Unit 2's.

TABLE 5.2-5 (Sheet 1 of 2)

STEAM GENERATOR DESIGN DATA*

(Model 51)

	~~Unit 1~~	~~Unit 2~~
Number of Steam Generators	~~4~~	4
Design Pressure (Reactor coolant/steam), psig	~~2485/1085~~	2485/1085
Reactor Coolant Hydrostatic Test Pressure (tube side-cold), psig	~~3107~~	3107
Design Temperature (reactor coolant/steam), °F	~~650/600~~	650/600
Reactor Coolant Flow, lb/hr	~~33.53×10^6~~	33.05×10^6 ~~33.47×10^6~~
Total Heat Transfer Surface Area, ft^2	~~51,500~~	51,500
Heat Transferred, Btu/hr	~~2857×10^6~~	2920×10^6
Steam Conditions at Full Load, Outlet Nozzle:		
Steam Flow, lb/hr	~~3.61×10^6~~	3.74×10^6
Steam Temperature, °F	~~519~~	519
Steam Pressure, psig	~~805~~	805
Maximum Moisture Carryover, wt percent	~~0.25~~	0.25
Feedwater, °F	~~435~~	435
Overall Height, ft-in.	~~67-8~~	67-8
Shell OD (upper/lower), in.	~~175-3/4 / 135~~	175-3/4 / 135
Number of U-tubes	~~3388~~	3388
U-tube OD, in.	~~0.875~~	0.875
Tube Wall Thickness (minimum), in.	~~0.050~~	0.050
Number of Manways/ID in.	~~4/16~~	4/16
Number of handholes/ID, in.	~~2/6~~	2/6

*Quantities are for each steam generator

TABLE 5.2-5 (Sheet 2 of 2)

STEAM GENERATOR DESIGN DATA*

(Model 51)

	~~Unit 1~~ Rated Load	~~Unit 2~~ No Load
Reactor Coolant Water Volume, ft^3	1080	1080
Primary Side Fluid Heat Content, Btu	28.7×10^6	27.7×10^6
Secondary Side Water Volume, ft^3	1838	3524
Secondary Side Steam Volume, ft^3	4030	2344
Secondary Side Steam Fluid Heat Content, Btu	5.738×10^7	9.628×10^7

*Quantities are for each steam generator

TABLE 5.2-7

REACTOR COOLANT PIPING DESIGN PARAMETERS

	~~Unit 1~~	~~Unit 2~~
Reactor Inlet Piping ID, in.	~~27-1/2~~	27-1/2
Reactor Inlet Piping Nominal Thickness, in.	~~2.38~~	2.38
Reactor Outlet Piping ID, in.	~~29~~	29
Reactor Outlet Piping Nominal Thickness, in.	~~2.50~~	2.50
Coolant Pump Suction Piping ID, in.	~~31~~	31
Coolant Pump Suction Piping Nominal Thickness, in.	~~2.66~~	2.66
Pressurizer Surge Line Piping ID, in.	~~11.500~~	(1)
Pressurizer Surge Line Piping nominal Thickness, in.	~~1.25~~	(2)
Design/Operating Pressure, psig	~~2485/2235~~	2485/2235
Hydrostatic Test Pressure (Cold), psig	~~3107~~	3107
Design Temperature, °F	~~650~~	650
Design Temperature (pressurizer surge line), °F	~~680~~	680
Water Volume, (all 4 loops including surge line) ft^3	~~2455~~	1455
Design Pressure (pressurizer relief lines), psig	~~(1)~~	(3)
Design Temperature (pressurizer relief lines), °F	~~(1)~~	(3)

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Editorial note: The total water volume originally reported for unit 1, 2455 ft^3, was a typographical error. The correct value is 1455 ft^3, the same as unit 2's

(1) Unit 1 11.188, Unit 2 11.500 inches
(2) Unit 1 1.25, Unit 2 1.460

(3) From pressurizer to safety valve 2485 psig 650°F
From safety valve to pressurizer relief tank 600 psig 600°F.

TABLE 5.5-1 (Sheet 1 of 3)

RESIDUAL HEAT REMOVAL SYSTEM DESIGN PARAMETERS

Code Requirements

Residual Heat Exchangers (Tube Side)	ASME III, Class C
(Shell Side)	ASME VIII
Residual Heat Removal Piping and Valves	ANSI B31.1.0* ANSI B31.7**

General

Plant design life, years	40
Component cooling water supply temperature design, °F	95
Reactor coolant temperature at startup of decay heat removal °F	350
Time to cool Reactor Coolant System from 350°F to 140°F, starting at 4 hours after shutdown, hr	16
Refueling water storage temperature, °F	Ambient
Decay heat generation at 20 hours after shutdown, Btu/hr	~~70.6 x 10^6 (Unit No. 1)~~ 72.1 x 10^6 ~~(Unit No. 2)~~
H_3BO_3 concentration in refueling water storage tank, ppm boron	~2000

* Used for design.

** For piping not supplied by the NSSS supplier, material inspection fabrication and quality control conform to ANSI B31.7. Where not possible to comply with ANSI B31.7, the requirements of ASME III-1971, which incorporated ANSI B31.7, were adhered to.

10.2 TURBINE GENERATOR

10.2.1 DESIGN BASES

The Steam and Power Conversion System is designed to convert the heat produced in the reactor to electrical energy. Heat absorbed by the Reactor Coolant System is transferred to the feedwater in four steam generators. The feedwater system provides sufficient feedwater flow to the four steam generators where removal of heat from the Reactor Coolant System results in sufficient steam formation to drive the turbine generator units as follows:

	No. 1 Unit	No. 2 Unit
AT 100% REACTOR POWER ~~Maximum Guaranteed Rating~~		
Gross Output, Mwe	1158 ~~1132~~	1158
Anticipated Net Output, Mwe	~~1090~~ 1115	1115
Maximum Calculated Load		
Gross Output, Mwe	1176	1201
Anticipated Net Output, Mwe	1130	1155

10.2.2 SYSTEM DESCRIPTION

10.2.2.1 Turbine-Generator

The turbine is a four-casing, tandem-compound, six flow exhaust, 1800 rpm unit with 44-inch long last stage blades. The turbine shaft is directly connected to the ac generator. A brushless exciter is coupled to the generator. The generator is hydrogen cooled with water-cooled stator windings. It is rated at 1,300,000 KVA at 75 psig hydrogen pressure, 0.90 PF, 0.48 SCX, 3 phase, 60 cps, 25 KV, and 1800 rpm. Generator

ANSI-B31.7, Nuclear Power Piping. Where not possible to comply with ANSI B31.7, the requirements of ASME III-1971, which incorporated ANSI B31.7, were adhered to.

(b) Principal System Valves:

Main Steam Safety Valves - ASME Boiler and Pressure Vessel Code, Section III, Class A.

Main Steam Relief Valves - ASME Boiler and Pressure Vessel Code, Section III, Class II (Class I for materials, inspections, fabrication and quality control).

Main Steam Stop Valves - ASME Boiler and Pressure Vessel Code, Section III, Class II (Class I for materials, inspections, fabrication and quality control).

Feedwater Isolation Valves - ASME Boiler and Pressure Vessel Code, Section III, Class II (Class I for materials, inspections, fabrication and quality control).

10.3.2 SYSTEM DESCRIPTION

10.3.2.1 Main Steam System

The Main Steam System is shown in Figure 10.3-1.

The Main Steam System for each unit conveys saturated steam from four steam generators to the high pressure turbine with less than 40 psi pressure drop. The steam conditions at full load of approximately 3,740,000 pounds per hour of 750 psig 513°F steam were ~~Though each steam generator for the No. 1 Unit is designed to furnish approximately 3,600,000 pounds per hour of 750 psig, 513 F steam to the turbine, the higher design flow rate for the No. 2 Unit, approximately 3,700,000 pounds per hour to the turbine for each steam generator, was~~ used for the system design of both units. Reheat is