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December 16, 1983

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Byron Generating Station Units 1 and 2
Braidwood Generating Station Units 1 and 2
Steam Generator Tube Vibration
NRC Docket Nos. 50-454, 50-455, 50-456, and 50-457

References (a): February 9, 1983 letter from T. R. Tramm to H. R. Denton.

(b): April 12, 1983 letter from T. R. Tramm to H. R. Denton.

(c): July 18, 1983 NRC Summary of Meeting on July 7, 1983 with the Technical Review Committee.

(d): July 18, 1983 letter from L. D. Butterfield, Jr. to D. G. Eisenhut.

(e): August 1, 1983, letter from L. D. Butterfield, Jr. to D. G. Eisenhut.

Dear Mr. Denton:

This letter provides additional information regarding the changes being made to the Byron and Braidwood steam generators to minimize tube vibration. NRC review of this information should enable closure of Outstanding Item 10 of the Byron SER.

As described in previous generic correspondence meetings and hearings, selected steam generator tubes are being expanded at two tube support plates and 10% of the main feedwater flow is being diverted to the auxiliary feedwater nozzle. Extensive reviews and analyses of these modifications have been performed by Westinghouse and by the Counterflow Steam Generator Owners Review Group. These efforts have verified that the modifications will be effective in reducing tube vibration to acceptable levels and that the modifications will introduce no unacceptable safety consequences during normal or transient operating conditions.

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This letter provides the plant-specific information, details of this modification and the results of plant-specific safety analyses on the split-feed arrangement. Tables 2.5-1 and 2.5-2 of references (d) and (e) list the FSAR sections and design transients which have been reviewed. Special attention has been given to the prevention of waterhammer and to the impact of these changes upon plant operating procedures. We have concluded that the Byron and Braidwood plants can be operated safely with the modified steam generators and feedwater piping.

Attachment A to this letter contains revised FSAR pages. These contain the necessary revisions to the facility description and accident analyses. These pages will be incorporated into the FSAR at the earliest opportunity.

As indicated in references (d) and (e), inservice inspection plans and tube plugging criteria are also being addressed on a plant-specfic basis. For Byron and Braidwood, the Technical Specifications already proposed adequately cover these issues. The extensive inservice inspection program already agreed upon provides for the early detection of unanticipated problems with steam generator tubing. The 40% plugging criteria appears adequate to prevent mid-cycle failure of any tubes which are found to experience minor degradation. To provide additional assurance of the adequacy of the tube modifications, vibration measurements are to be made on selected tubes during operation of one of the first domestic units with expanded tubes. If these measurements indicate the need for additional inservice inspection, ISI changes can be easily incorporated into individual plant Technical Specifications such as Byron's.

Please address further questions regarding this matter to this office.

One signed original and fifteen copies of this letter and the enclosures are provided for NRC review.

Very truly yours,

T. R. Tramm

T. R. Tramm
Nuclear Licensing Administrator

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Enclosure

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ATTACHMENT A

FSAR Revisions to Implement 90/10

Feedwater Flow Split

1.	Table 3.6-12	Corrections
2.	Table 3.9-16	Revisions
3.	Table 4.1-1	Revisions and Corrections
4.	Table 4.4-1	Revisions and Corrections
5.	Figure 4.4-9	Revisions
6.	Page 5.1-4	Revisions
7.	Table 5.1-1	Revisions
8.	Figure 5.1-2	Revisions
9.	Section 5.4.2.5.3	Replacement Section
10.	Section 5.4.2.5.4	New Section
11.	Table 6.2-58	Revisions
12.	Section 10.4.7.3	Replacement Section
13.	Section 15.0.3.2	Revisions
14.	Figure 3.6-2	Revisions
15.	Figure 10.4-1	Revisions

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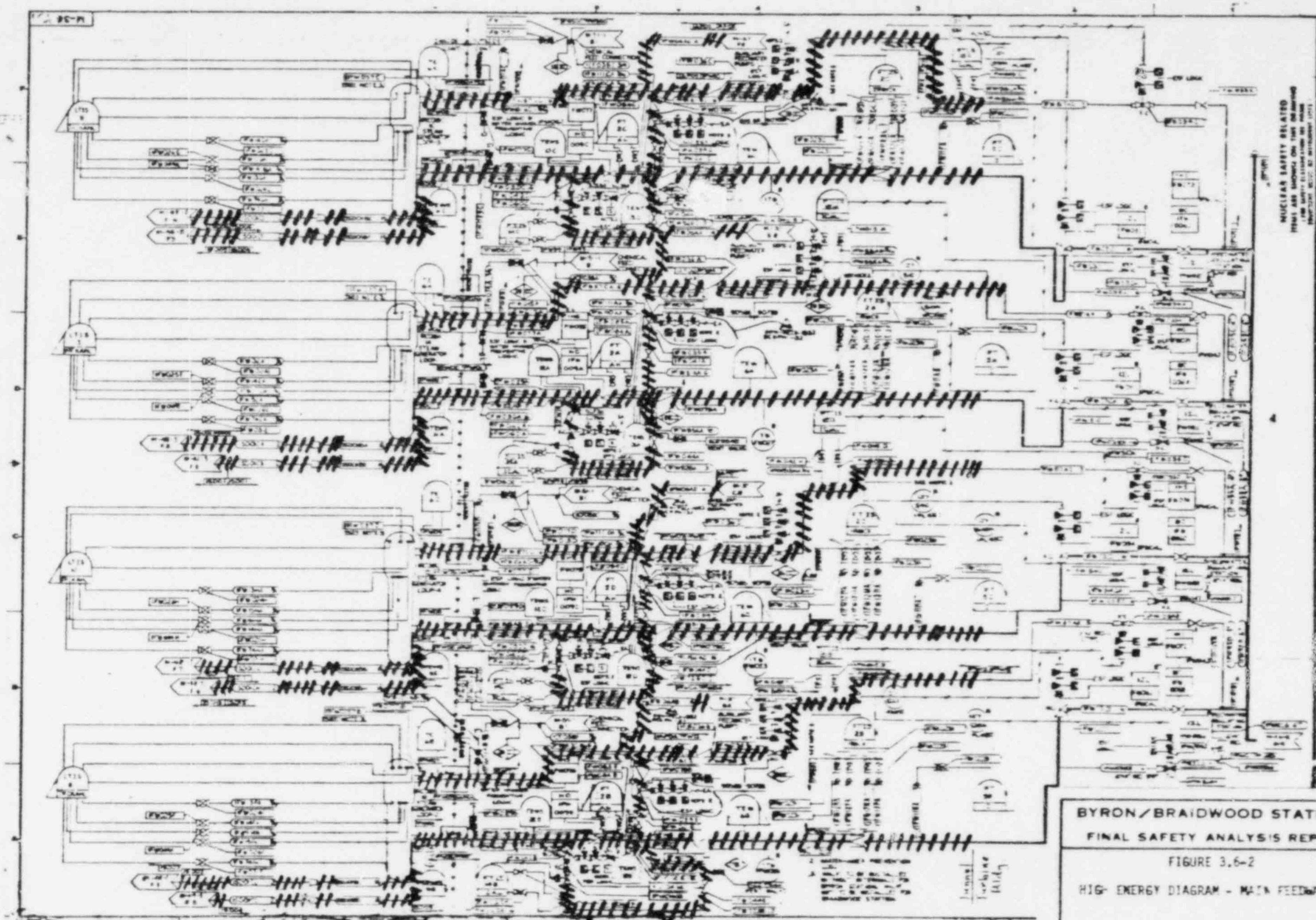
TABLE 3.6-12

CALCULATED STRESSES FOR POSTULATED BREAK POINTS

(For ASME Sec. III Class 2&3 and ANSI B31.1 Piping Systems)

PIPING SYSTEM LINE NUMBER(S)	BREAK ID	CALCULATED STRESS NORMAL & UPSET PLANT CONDITIONS (psi)	$0.8(1.2S_b+S_A)$ (psi)
FEEDWATER Loop 1			
1FW03DA-16"	B20A	17660	32400.
1FW03DA-16"	B20B	16832	32400.
1FW03DA-16"	B40A	13586	32400.
1FW03DA-16"	B65A	19150	32400.
1FW03DA-16"	B65B	20934	32400.
1FW03DA-16"	B80	15907	32400.
FEEDWATER Loop 2			
1FW03DB-16"	B5A	10706.	32400.
1FW03DB-16"	B30A	17847.	32400.
1FW03DB-16"	B30B	17293.	32400.
1FW03DB-16"	B55A	12908.	32400.
1FW03DB-16"	B85A	16971.	32400.
1FW03DB-16"	B85B	18565.	32400.
1FW03DB-16"	B100	14974.	32400.



BYRON/BRAIDWOOD STATIONS

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FIGURE 3.6-2

HIGH ENERGY DIAGRAM - MAIN FEEDWATER

TABLE 3.9-16 (Cont'd)

TAG NUMBER	ACTUATED BY	SIZE (in.)	BODY TYPE	QUALITY GROUP	ISI CATEGORY	P&ID
1CV8378A,B	--	3	Check	A	C	M-64-5
1CV9379A,B	--	3	Check	A	C	M-64-5
1CV8381	--	3	Check	B	C	M-64-5
1CV8442	--	2	Check	B	C	M-64-4
1CV8481A,B	--	4	Check	B	C	M-64-3
1CV8546	--	8	Check	B	C	M-64-4
1CV8804A	Motor	8	Gate	B	B	M-64-4
1FC009	--	4	Plug	B	A	M-63-1
1FC010	--	4	Plug	B	A	M-63-1
1FP010	Air	4	Gate	B	B	M-52-1
1FP011	Air	4	Gate	B	B	M-52-1
1FW009A-D	Hydraulic	16	Gate	B	B	M-36-1
1FW015A-D	--	.75	Globe	B	B	M-36-1
1FW035A-D	Air	3	Gate	B	B	M-36-1
1FW036A-D	--	3	Check	B	C	M-36-1
1FW039A-D	Air	6	Gate	B	B	M-36-1
1FW043A-D	Air	1	Globe	B	B	M-36-1
1FW078A-D	--	6	Check	B	C	M-36-1
1IA065	Air	3	Globe	B	A	M-55-2
1IA066	Air	1	Globe	B	A	M-55-2
1IA088	--	.5	Globe	B	A	M-55-2
1IA091	--	.75	Check	B	A	M-55-2
1MS001A-D	Comp. Air	32.75	Gate	B	A	M-35-1
1MS013A-D	--	6	Relief	B	C	M-35-1
1MS014A-D	--	6	Relief	B	C	M-35-1
1MS015A-D	--	6	Relief	B	C	M-35-1
1MS016A-D	--	6	Relief	B	C	M-35-1
1MS017A-D	--	6	Relief	B	C	M-35-1
1MS018A-D	Comp. Air	8	Relief	B	C	M-35-1
1MS021A-D	--	3	Globe	B	A	M-35-1
1MS101A-D	Air	4	Gate	B	A	M-35-1
0ØG059	Motor	3	Butterfly	B	C	M-47-2
0ØG060	Motor	3	Butterfly	B	C	M-47-2
0ØG061	Motor	3	Butterfly	B	C	M-47-2
0ØG062	Motor	3	Butterfly	B	C	M-47-2
0ØG063	Motor	3	Butterfly	B	C	M-47-2
0ØG064	Motor	3	Butterfly	B	C	M-47-2
1ØG057A	Motor	3	Butterfly	B	A	M-47-2
1ØG079	Motor	3	Butterfly	B	A	M-47-2
1ØG080	Motor	3	Butterfly	B	A	M-47-2
1ØG081	Motor	3	Butterfly	B	A	M-47-2
1ØG082	Motor	3	Butterfly	B	A	M-47-2
1ØG083	Motor	3	Butterfly	B	A	M-47-2
1ØG084	Motor	3	Butterfly	B	A	M-47-2
1ØG085	Motor	3	Butterfly	B	A	M-47-2
1PR001A,B	Air	1	Globe	B	A	M-78-10
1PR031	Air	1	Globe	B	A	M-78-10

TABLE 4.1-1

REACTOR DESIGN COMPARISON TABLE

THERMAL AND HYDRAULIC DESIGN PARAMETERS	BYRON AND BRAIDWOOD UNITS 1 and 2 (LOW PARASITIC FUEL)	BYRON AND BRAIDWOOD UNITS 1 and 2 (OPTIMIZED FUEL)	WCAP-9500 REFERENCE DESIGN
1. Reactor Core Heat Output, (100%), MW_t	3411	3411	3411
2. Reactor Core Heat Output, 10^6 Btu/hr	11641.7	11641.7	11641.7
3. Heat Generated in Fuel, %	97.4	97.4	97.4
4. Core Pressure, Nominal, psia(1)	2250	2280	2280
5. System Pressure, Minimum Steady State, psia(1)	2220	2250	2250
6. Minimum DNBR at Nominal Conditions			
Typical Flow Channel	2.09	2.47	2.40
Thimble (Cold Wall) Flow Channel	1.74	2.32	2.26
7. Minimum DNBR for Design Transients			
Typical Flow Channel	$\geq$1.30	>1.49	>1.49
Thimble Flow Channel	$\geq$1.30	>1.47	>1.47
8. DNB Correlation	"R" (W-3 with Modified Spacer Factor)	WRB-1(2)	WRB-1(2)
COOLANT FLOW			
9. Total Thermal Flow Rate, 10^5 lb_m/hr	138.6	144.9	143.3
10. Effective Flow Rate for Heat Transfer, 10^6 lb_m/hr	132.4	138.7	134.7
11. Effective Flow Area for Heat Transfer, ft^2	51.1	54.1	54.1
12. Average Velocity Along Fuel Rods, ft/sec	16.4	16.2	15.8
13. Average Mass Velocity, 10^6 lb_m/hr-ft^2	2.59	2.56	2.49

(1) Values used for thermal hydraulic core analysis

(2) The W-3 correlation is used for analysis of some accidents involving depressurization of the steam system. (See Table 15.0-2, sheet 1)

TABLE 4.1-1 (Cont'd)

THERMAL AND HYDRAULIC DESIGN PARAMETERS	BYRON AND BRAIDWOOD UNITS 1 and 2 (LOW PARASITIC FUEL)	BYRON AND BRAIDWOOD UNITS 1 and 2 OPTIMIZED FUEL	WCAP-9500 REFERENCE DESIGN
COOLANT TEMPERATURE, °F			
14. Nominal Inlet	556.9	559.2	561.6
15. Average Rise in Vessel	61.1	58.4	58.5
16. Average Rise in Core	63.6	60.7	61.8
17. Average in Core	590.4	591.1	594.2
18. Average in Vessel	587.4	588.4	592.3
HEAT TRANSFER			
19. Active Heat Transfer, Surface Area, ft^2	59,700	57,500	57,500
20. Average Heat Flux, Btu/hr-ft^2	189,800	197,200	197,200
21. Maximum Heat Flux for Normal Operation, Btu/hr-ft^2	440,300	457,500	457,500
22. Average Linear Power, kW/ft	5.44	5.44	5.44
23. Peak Linear Power for Normal Operation, kW/ft(*)	12.6	12.6	12.6
24. Peak Linear Power Resulting from Overpower Transients/Operator Errors (assuming a maximum overpower of 118%), kW/ft(**)	18.0	18.0	18.0
25 Peak Linear Power for Prevention of Centerline Melt, kW/ft(***)	>18.0	>18.0	>18.0

* This limit is associated with the value of F_Q = 2.32
** See Subsection 4.3.2.2.6
*** See Subsection 4.4.2.11.6.

TABLE 4.1-1 (Cont'd)

THERMAL AND HYDRAULIC DESIGN PARAMETERS	BYRON AND BRAIDWOOD UNITS 1 and 2 (LOW PARASITIC FUEL)	BYRON AND BRAIDWOOD UNITS 1 and 2 (OPTIMIZED FUEL)	WCAP-9500 REFERENCE DESIGN
26. Power Density, kW per Liter of Core(+)	104.5	104.5	104.5
27. Specific Power, kW per kg Uranium	38.4	41.9	41.9
FUEL CENTRAL TEMPERATURE			
28. Peak at Peak Linear Power for Prevention of Centerline Melt, $^{\circ}F$	4700	4700	4700
29. Pressure Drop(++)			
Across Core, psi	26.9$\pm$2.7(+++)	26.3$\pm$2.6	25.7$\pm$2.6
Across Vessel, Including Nozzle psi	47.4$\pm$4.7(+++)	46.4$\pm$4.6	45.7$\pm$4.6
CORE MECHANICAL DESIGN PARAMETERS			
30. Design	RCC Canless 17 x 17	RCC Canless 17 x 17	RCC Canless 17 x 17
31. Number of Fuel Assemblies	193	193	193
32. UO_2 Rods per Assembly	264	264	264
33. Rod Pitch, in.	0.496	0.496	0.496
34. Overall Dimensions, in.	8.426 x 8.426	8.426 x 8.426	8.426 x 8.426
35. Fuel Weight (as UO_2), lb	222,739	204,236	204,236
36. Clad Weight, lb	50,913	43,376	43,376

+ Based on cold dimensions and 95% of theoretical density fuel
++ Based on best estimate reactor flow rate as discussed in Section 5.1
+++ Pressure drops revised based on results from Reference 2.

TABLE 4.1-1 (Cont'd)

THERMAL AND HYDRAULIC DESIGN PARAMETERS	BYRON AND BRAIDWOOD UNITS 1 and 2 (LOW PARASITIC FUEL)	BYRON AND BRAIDWOOD UNITS 1 and 2 (OPTIMIZED FUEL)	WCAP-9500 REFERENCE DESIGN
37. Number of Grids per Assembly	8 - Type R	2-Type R, 6-Type Z	2-Type R, 6-Type Z
38. Composition of Grids	Inconel 718	2 End Grids - Inconel 718 6 Intermediate Grids - Zircaloy 4	2 End Grids - Inconel 718 6 Intermediate Grids - Zircaloy 4
39. Loading Technique	3 Region Nonuniform	3 Region Nonuniform	3 Region Nonuniform
CORE MECHANICAL DESIGN PARAMETERS			
FUEL RODS			
40. Number	50,952	50,952	50,952
41. Outside Diameter, in.	0.374	0.360	0.360
42. Diametral Gap, in.	0.0065	0.0062	0.0062
43. Cladding Thickness, in.	0.0225	0.0225	0.0225
44. Cladding Material	Zircaloy-4	Zircaloy-4	Zircaloy-4
FUEL PELLETS			
45. Material	UO_2 Sintered	UO_2 Sintered	UO_2 Sintered
46. Density (% of Theoretical)	95	95	95
47. Diameter, in.	0.3225	0.3088	0.3088
48. Length, in.	0.530	0.507	0.507

TABLE 4.4-1

THERMAL AND HYDRAULIC COMPARISON TABLE

DESIGN PARAMETERS	BYRON AND BRAIDWOOD UNITS 1 AND 2 LOW PARASITIC FUEL	WCAP 9500 REFERENCE DESIGN	BYRON AND BRAIDWOOD UNITS 1 AND 2 (OPTIMIZED FUEL)
Reactor Core Heat Output (100%), MWt	3411	3411	3411
Reactor Core Heat Output, 10^6 Btu/hr	11641.7	11641.7	11641.7
Heat Generated in Fuel, %	97.4	97.4	97.4
System Pressure, Nominal, psia(2)	2250	2280	2280
System Pressure, Minimum Steady-State, psia(2)	2220	2250	2250
Minimum DNBR at Nominal Conditions(1)			
Typical Flow Channel	2.09	2.40	2.47
Thimble (Cold Wall) Flow Channel	1.74	2.26	2.32
Minimum DNBR for Design Transients(1)			
Typical Flow Channel	$\geq$1.30	$\geq$1.49	$\geq$1.49
Thimble Flow Channel	$\geq$1.30	$\geq$1.47	$\geq$1.47
DNB Correlation	"R" (W-3 with Modified Spacer Factor)	WRB-1	WRB-1
COOLANT FLOW			
Total Thermal Flow Rate, 10^6 lb_m/hr	138.6	143.3	144.9
Effective Flow Rate for Heat Transfer, 10^6 lb_m/hr	132.4	134.7	138.7
Effective Flow Area for Heat Transfer, ft^2	51.1	54.1	54.1
Average Velocity Along Fuel Rods, ft/sec	16.4	15.8	16.2
Average Mass Velocity, 10^6 lb_m/hr-ft^2	2.59	2.49	2.56

TABLE 4.4-1 (Continued)

THERMAL AND HYDRAULIC COMPARISON TABLE

DESIGN PARAMETERS	BYRON AND BRAIDWOOD UNITS 1 AND 2 LOW PARASITIC FUEL	WCAP 9500 REFERENCE DESIGN	BYRON AND BRAIDWOOD UNITS 1 AND 2 (OPTIMIZED FUEL)
COOLANT TEMPERATURE			
Nominal Inlet, °F	556 9	561.6	559.2
Average Rise in Vessel, °F	61.1	58.5	58.4
Average Rise in Core, °F	63.6	61.[illegible]	60.7
Average in Core, °F	59[illegible].4	594.2	591.1
Average in Vessel, °F	587.4	592.[illegible]	588.4
HEAT TRANSFER			
Active Heat Transfer, Surface Area, ft^2	59,700	57,500	57,500
Average Heat Flux, Btu/hr-ft^2	189,800	197,200	197,200
Maximum Heat Flux for Normal Operation, Btu/hr-ft^2	440,300	457,500	457,500
Average Linear Power, kW/ft	5.44	5.44	5.44
Peak Linear Power for Normal Operation, kW/ft(*)	12.6	12.6	12.6
Peak Linear Power Resulting from Overpower Transients /Operator Errors (assuming a maximum overpower of 118%), kW/ft(**)	18.0	18.0	18.0
Peak Linear Power for Prevention of Centerline Melt, kW/ft(***)	>18.0	>18.0	>18.0
Power Density, kW per liter of core(+)	104.5	104.5	104.5
Specific Power, kW per kg Uranium(+)	38.4	41.9	41.9

TABLE 4.4-1 (Continued)

THERMAL AND HYDRAULIC COMPARISON TABLE

DESIGN PARAMETERS	BYRON AND BRAIDWOOD UNITS 1 AND 2 LOW PARASITIC FUEL	WCAP 9500 REFERENCE DESIGN	BYRON AND BRAIDWOOD UNITS 1 AND 2 (OPTIMIZED FUEL)
FUEL CENTRAL TEMPERATURE			
Peak at Peak Linear Power for Prevention of Centerline Melt, °F	4700	4700	4700
Pressure Drop(++)			
Across Core, psi	26.9 ± 2.7+++	25.7 ± 2.6	26.3 ± 2.6
Across Vessel, including nozzle psi	47.4 ± 4.7+++	45.7 ± 4.6	46.4 ± 4.6

* This limit is associated with the value of F_Q = 2.32
** See Subsection 4.3.2.2.6.
*** See Subsection 4.4.2.11.6.
\+ Based on cold dimensions and 95% of theoretical density fuel
++ Based on best estimate reactor flow rate as discussed in Section 5.1
+++ Pressure Drops updated based on results from Reference 5.
(1) These numbers are not directly comparable for each plant design due to the incorporation of a different thermal design procedure and DNB correlation in the present core.
(2) Value used for thermal hydraulic core analysis.

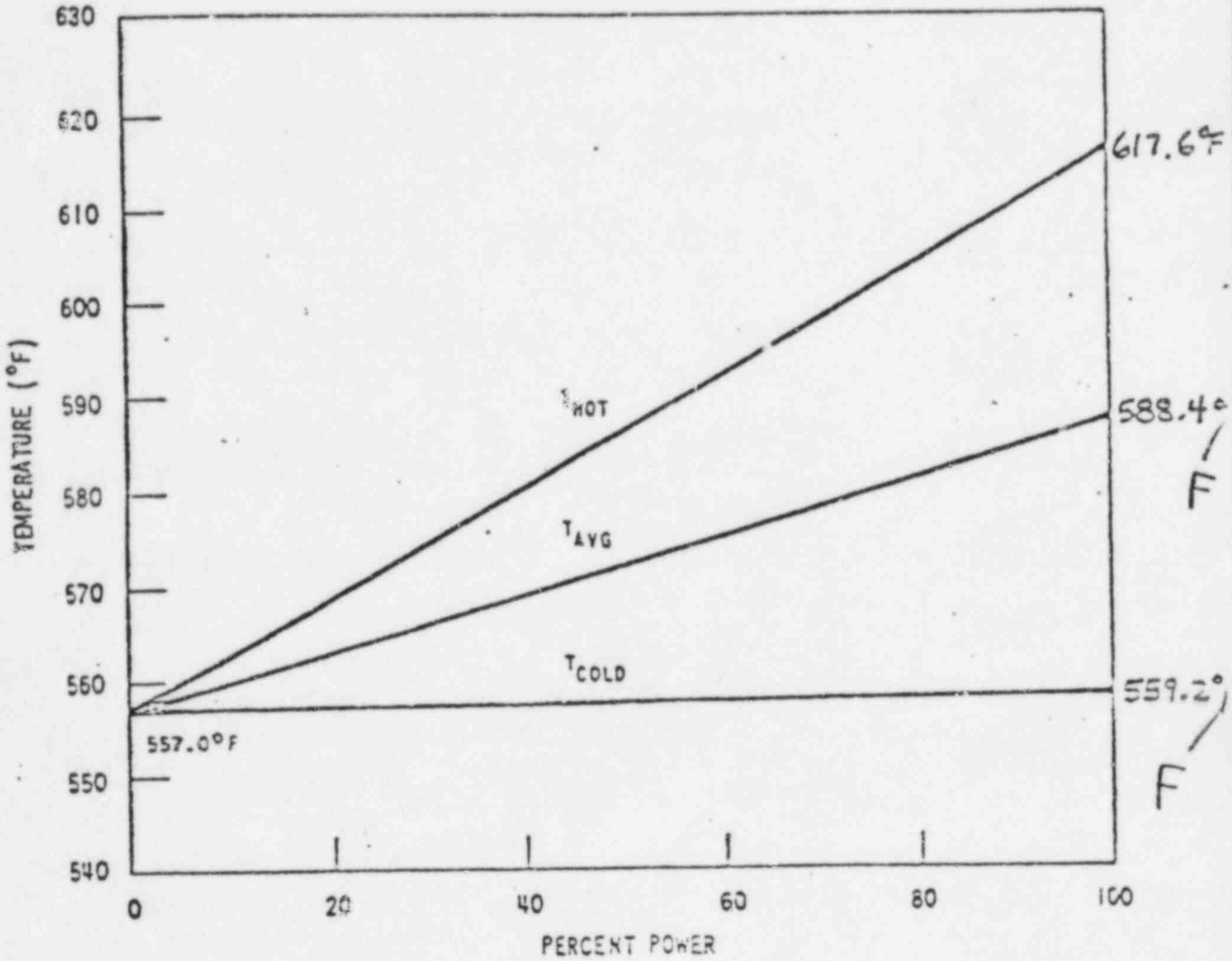

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Figure 4.4-9.
Reactor Coolant System Temperature
Percent Power Map

operating plants have indicated that the actual flow has been well above the flow specified for the thermal design of the plant. By applying the design procedure described in the following, it is possible to specify the expected operating flow with reasonable accuracy.

Three reactor coolant flow rates are identified for the various plant design considerations. The definitions of these flows are presented in the following paragraphs.

Best Estimate Flow

The best estimate flow is the most likely value for the actual plant operating condition. This flow is based on the best estimate of the reactor vessel, steam generator and piping flow resistance, and on the best estimate of the reactor coolant pump head-flow capacity, with no uncertainties assigned to either the system flow resistance or the pump head. System pressure drops, based on best estimate flow, are presented in Table 5.1-1. Although the best estimate flow is the most likely value to be expected in operation, more conservative flow rates are applied in the thermal and mechanical designs.

Thermal Design Flow

Thermal design flow is the basis for the reactor core thermal performance, the steam generator thermal performance, and the nominal plant parameters used throughout the design. To provide the required margin, the thermal design flow accounts for the uncertainties in reactor vessel, steam generator and piping flow resistances, reactor coolant pump head, and the methods used to measure flow rate. The thermal design flow is approximately 5.9% less than the best estimate flow. The thermal design flow is confirmed when the plant is placed in operation. Tabulations of important design and performance characteristics of the reactor coolant systems, as provided in Table 5.1-1, are based on the thermal design flow.

Mechanical Design Flow

Mechanical design flow is the conservatively high flow used in the mechanical design of the reactor vessel internals and fuel assemblies. To ensure that a conservatively high flow is specified, the mechanical design flow is based on a reduced system resistance and on increased pump head capability. The mechanical design flow is approximately 3.7% greater than the best estimate flow. Maximum pump overspeed results in a peak reactor coolant flow of 120% of the mechanical design flow. This overspeed condition, which is coincident with a turbine-generator overspeed of 20%, is only applicable if, when a turbine trip would be actuated, the turbine governor fails and the turbine is tripped by the mechanical overspeed trip device.

TABLE 5.1-1

SYSTEM DESIGN AND OPERATING PARAMETERS

Plant design life, years	40
Nominal operating pressure, psia	2250
Total system volume including pressurizer and surge line, ft^3	12,074
System liquid volume, including pressurizer water at maximum guaranteed power, ft^3	11,695
Pressurizer spray rate, gpm	900
Pressurizer heater capacity, kW	1800
Pressurizer relief tank volume, ft^3	1800

SYSTEM THERMAL AND HYDRAULIC DATA
(Based on Thermal Design Flow)

	4 PUMPS RUNNING	3 PUMPS** RUNNING
NSSS power, MWt	3425	2569
Reactor power, MWt	3411	2560
Thermal design flows, gpm*		
Active loop	94,400	98,000
Idle loop	--	0
Reactor	377,600	294,000
Total reactor flow, 10^6 lb/hr	140.3	110.5
Temperatures, °F		
Reactor vessel outlet	618.4	612.2
Reactor vessel inlet	558.4	552.3

TABLE 5.1-1 (Cont'd)

Steam generator outlet	558.1	552.1
Steam generator steam	543.3	538.0
Feedwater	440	408.0
Steam pressure, psia	990	947
Total steam flow, 10^6 lb/hr	15.13	10.84
Best estimate flows, gpm*		
Active loop	100,300	105,300
Idle loop	--	0
Reactor	401,200	315,900
Mechanical design flows, gpm*		
Active loop	104,000	109,500
Idle loop	--	0
Reactor	416,000	328,500

SYSTEM PRESSURE DROPS
(Based on Four-Loop Best Estimate Flow)

Reactor vessel ΔP	44.7
Steam generator ΔP, psi	38.3
Hot let piping Δp, psi	2.3
Pump suction piping ΔP, psi	3.2
Cold leg piping ΔP, psi	2.3
Pump head, feet	290

*At pump discharge.

**Calculated assuming all feedwater enters steam generators through the main feedwater nozzle.

NOTES TO FIGURE 5.1-2

MODE A STEADY-STATE FULL POWER OPERATION

LOCATION	FLUID	PRESSURE PSIG	TEMPERATURE °F	FLOW GPM(1)	FLOW LB/HR(2)	VOLUME
1	R.C.	2235.0	617.9	112,259	37.36	-
2	"	2233.1	617.9	112,159	37.33	-
3	"	2195.9	556.7	100,300	37.33	-
4	"	2192.4	556.7	100,494	37.40	-
5	"	2285.1	556.9	100,400 (5)	37.40	-
6	"	2283.2	556.9	100,300	37.36	-
7(3)	"	2234.1	617.9	100	0.0333	-
8(4)	"	2285.1	556.9	100	0.0371	-
9	"	2194.2	587.0	199	0.0704	-
10-18	"	SEE LOOP #1 SPECIFICATIONS				-
19-27	"	SEE LOOP #1 SPECIFICATIONS				-
28-36	"	SEE LOOP #1 SPECIFICATIONS				-
37	"	2285.1	556.9	1.0	0.0004	
38	"	2285.1	556.9	1.0	0.0004	
39	"	2235.0	556.9	2.0	0.0008	
40	STEAM	2235.0	652.7	-	-	720
41	R.C.	2235.0	652.7	-	-	1080
42	"	2235.0	652.7	2.5	0.0008	-
43	"	2235.0	652.7	2.5	0.0008	-
44	STEAM	2235.0	652.7	0	0	-
45	R.C.	2235.0	<652.7	0	0	MINIMIZE
46	N_2	3.0	120	0	0	-
47	R.C.	2235.0	<652.7	0	0	MINIMIZE
48	N_2	3.0	120	0	0	-
49	"	3.0	120	0	0	-
50	"	3.0	120	-	-	450
51	PRT WATER	3.0	120	-	-	1350

(1) At the conditions specified.

(2) X 10^6

(3) Location point refers to the three 1" connections on the hot leg.

(4) Location point refers to the 2" connection on the cold leg.

(5) Most recent calculation of best estimate flow is 100,300 gpm. Values on this table would only be minimally affected if recalculated using latest best estimate flow.

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FIGURE 5.1-2

REACTOR COOLANT SYSTEM PROCESS FLOW DIAGRAM

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5.4.2.5.3 Mechanical and Flow-Induced Vibration Under Normal Operation

In the design of Westinghouse steam generators, the potential for tube wall degradation attributable to mechanical or flow-induced excitation has been thoroughly evaluated. The evaluation included detailed analyses of the tube support systems for various mechanisms of tube vibration.

The primary cause of tube vibration in heat exchangers is hydrodynamic excitation due to secondary fluid flow on the outside of the tubes. In the range of normal steam generator operating conditions, the effects of primary fluid flow inside the tubes and mechanically induced tube vibration are considered to be negligible.

To evaluate flow induced tube vibration in the preheater region of the tube bundle, Westinghouse undertook an extensive program employing data from operating plants, full and partial scale model tests, and analytical tube vibration models. Operating plant data consisted of tube wear data from pulled tube evaluations and eddy current tests and tube motion data from accelerometers installed inside selected tubes. Model testing generated tube wear data, flow velocity distributions, tube motion parameters, and flow-induced tube vibration forcing functions. The tube vibration analyses applied the forcing functions to produce tube motion data. The results of this evaluation were consistent with the early operating experience of preheat steam generators.

On the basis of an extensive model test and analysis program, Westinghouse designed, verified, and implemented a modification to the steam generator to reduce tube vibratory response to preheater inlet flow excitation. Additionally, the magnitude of the flow forcing function was reduced through implementation of a preheater flow bypass arrangement in the feedwater system. The verification of the performance of the modifications in reducing tube excitation and response was done with input from a full-scale test under simulated conservative flow and tube support conditions.

Fatigue of the tubes in the preheater region which are subject to flow-induced excitation is not a concern since the maximum resultant stresses in the tube are below the endurance limit of the material.

For areas of the tube bundle other than the preheater, parallel flow analyses were performed to determine the vibratory deflections. These analyses indicate that the flow velocities are sufficiently low such that they result in negligible fatigue and vibratory amplitudes. The support system, therefore, is deemed adequate with regard to parallel flow excitation.

To evaluate crossflow at the exit of the downcomer flow to the tube bundle and at the top of the bundle in the U-bend area, Westinghouse performed an experimental research program of crossflow in tube arrays with the specific parameters of the steam generator. Air and water model tests were employed. The results of this research indicate that these regions of the bundle are not subject to the vortex shedding mechanism of tube excitation. Vortex shedding was found not to be a significant mechanism in these two regions for the following reasons:

a. Flow turbulence in the downcomer and tube bundle inlet region inhibit the formation of Von Karman vorticies.

b. Both axial and crossflow velocity components exist on the tubes. The axial flow component disrupts the Von Karman vortices.

This research program was also the basis for evaluation of the fluid-elastic mechanism due to cross flow at the tubesheet. The evaluation showed the adequacy of the tube support arrangement.

Flow turbulence can result in some tube excitation in these regions. This excitation is of little concern, however, since:

a. Maximum stresses in the tubes are at least an order of magnitude below the fatigue endurance limit of the tube material, and

b. Tube support arrangements preclude significant vibratory motion.

In summary, tube vibration has been thoroughly evaluated. Mechanical and primary flow excitation are considered negligible. Secondary flow excitation has been evaluated. From this evaluation, it is concluded that if tube vibration does occur, the magnitude will be limited. Tube fatigue due to the vibration is judged to be negligible. Any tube wear resulting from the tube vibration would be limited and would progress slowly. This allows use of a periodic tube inservice inspection program for detection and followup of any tube wear. This inservice inspection program, in conjunction with tube plugging criteria, provides for safe operation of the steam generators.

5.4.2.5.4 Allowable Tube Wall Thinning Under Accident Conditions

An evaluation is performed to determine the extent of tube wall thinning that can be tolerated under accident conditions. Under such a postulated design-basis accident, vibration is of short enough duration that there is no endurance problem. The results of a study made on "D series" (.75 inch nominal diameter .043 inch nominal thickness) tubes under accident loading are discussed in WCAP-7832 (Reference 3) and show that a minimum wall thickness of .026 inches would have a maximum faulted condition stress (i.e., due to combined LOCA and safe shutdown earthquake loads) that is less than the allowable limit. This thickness is .010 inches less than the minimum steam generator tube wall thickness .039 reduced to .036 inches by the assumed general corrosion and erosion loss of .0033 inches.

The corrosion rate is based on a conservative weight loss rate for Inconel tubing in flowing 650° F primary side reactor coolant fluid. The weight loss, when equated to a thinning rate and projected over a 40-year plant life with appropriate reduction after initial hours, is equivalent to .083 mils thinning. The assumed corrosion rate of 3 mils leaves a conservative 2.917 mils for general corrosion thinning on the secondary side.

The steam generator tubes, existing originally at their minimum wall thickness and reduced by a very conservative general corrosion loss, still provide quite an adequate safety margin. Thus, it can be concluded that the ability of the steam generator tubes to withstand accident loadings is not affected by a lifetime of general corrosion losses.

TABLE 6.2-58 (Cont'd)

1 SYSTEM NAME	2 GDC REQUIREMENT MET	3 PENETRATION NUMBER	4 FLUID	5 LINE SIZE (INCHES)	6 ESSENTIAL**	7 REFERENCE DRAWING	8 ISOLATION VALVE NUMBER	9 VALVE LOCATION (INSIDE OR OUTSIDE CONTAINMENT)	10 TYPE C LEAK TEST (YES OR NO)	11 DISTANCE TO OUTER-MOST ISOLATION VALVE (FEET)	12 VALVE TYPE	13 VALVE OPERATOR	14 NORMAL POSITION	15 SHUTDOWN POSITION	16 POST ACCIDENT POSITION	17 POWER FAILURE POSITION	18 CLOSURE TIME (SECS.)	19 ISOLATION SIGNALS	20 PRIMARY MODE OF ACTUATION	21 SECONDARY MODE OF ACTUATION	22 POWER SOURCE	23 ISOLATION VALVE CONFIGURATION
	57	101	Water	4	YES	M-37	1AF013B	Outside	No	57.66	Globe	MO	Open	Closed	Open	As Is	23.8	FW	RM	M	1E	10
	57	101	Water	4	YES	M-37	1AF013F	Outside	No	53.0	Globe	MO	Open	Closed	Open	As Is	23.7	FW	RM	M	1E	10
	57	101	Water	3/4		M-36-1	1FW015B	Outside	No	46.75	Globe	M	Closed	Closed	Closed	N/A	N/A	N/A	M	M	N/A	14
	57	87	Water	16	YES	M-36-1	1FW009C	Outside	No	13.75	Gate	HO	Open	Closed	Closed	Closed	5.0	FW	A	RM	1E	10
	57	102	Water	4	YES	M-37	1AF013C	Outside	No	55.75	Globe	MO	Open	Closed	Open	As Is	23.9	FW	RM	M	1E	10
	57	102	Water	4	YES	M-37	1AF013G	Outside	No	52.25	Globe	MO	Open	Closed	Open	As Is	24.1	FW	RM	M	1E	10
	57	102	Water	3/4		M-36-1	1FW015C	Outside	No	46.75	Globe	M	Closed	Closed	Closed	N/A	N/A	N/A	M	M	N/A	14
	57	76	Water	16	YES	M-36-1	1FW009D	Outside	No	13.75	Gate	HO	Open	Closed	Closed	Closed	5.0	FW	A	RM	1E	10
	57	99	Water	4	YES	M-37	1AF013D	Outside	No	57.75	Globe	MO	Open	Closed	Open	As Is	23.8	FW	RM	M	1E	10
	57	99	Water	4	YES	M-37	1AF013H	Outside	No	54.25	Globe	MO	Open	Closed	Open	As Is	23.6	FW	RM	M	1E	10
	57	99	Water	3/4		M-36-1	1FW015D	Outside	No	46.75	Globe	M	Closed	Closed	Closed	N/A	N/A	N/A	M	M	N/A	14
	57	100	Water	3	YES	M-36-1	1FW035A	Outside	No	29.0	Globe	AO/S	Open	Closed	Closed	Closed	6.0	FW	A	RM	1E	11
	57	101	Water	3	YES	M-36-1	1FW035B	Outside	No	29.0	Globe	AO/S	Open	Closed	Closed	Closed	6.0	FW	A	RM	1E	11
	57	102	Water	3	YES	M-36-1	1FW035C	Outside	No	32.5	Globe	AO/S	Open	Closed	Closed	Closed	6.0	FW	A	RM	1E	11
	57	99	Water	3	YES	M-36-1	1FW035D	Outside	No	32.5	Globe	AO/S	Open	Closed	Closed	Closed	6.0	FW	A	RM	1E	11
	57	100	Water	6	YES	M-36-1	1FW039A	Outside	No	14.5	Gate	AO/S	Open	Closed	Closed	Closed	6.0	FW	A	RM	1E	11
	57	101	Water	6	YES	M-36-1	1FW039B	Outside	No	14.5	Gate	AO/S	Open	Closed	Closed	Closed	6.0	FW	A	RM	1E	11
	57	102	Water	6	YES	M-36-1	1FW039C	Outside	No	14.5	Gate	AO/S	Open	Closed	Closed	Closed	6.0	FW	A	RM	1E	11
	57	99	Water	6	YES	M-36-1	1FW039D	Outside	No	14.5	Gate	AO/S	Open	Closed	Closed	Closed	6.0	FW	A	RM	1E	11
	57	79	Water	3	YES	M-36-1	1FW043A	Outside	No	27.25	Globe	AO/S	Closed	Closed	Closed	Closed	6.0	FW	A	RM	1E	11
	57	84	Water	3	YES	M-36-1	1FW043B	Outside	No	27.25	Globe	AO/S	Closed	Closed	Closed	Closed	6.0	FW	A	RM	1E	11
	57	87	Water	3	YES	M-36-1	1FW043C	Outside	No	27.25	Globe	AO/S	Closed	Closed	Closed	Closed	6.0	FW	A	RM	1E	11
	57	76	Water	3	YES	M-36-1	1FW043D	Outside	No	27.25	Globe	AO/S	Closed	Closed	Closed	Closed	6.0	FW	A	RM	1E	11
Waste	56	47	Water	2	YES	M-48-6	1RF026	Inside	Yes	5.8	Plug	AO/S	Open	Open	Closed	Closed	15	T	A	RM	1E	2
Disposal	56	47	Water	2	YES	M-48-6	1RF027	Outside	Yes	4.6	Plug	AO/S	Open	Open	Closed	Closed	15	T	A	RM	1E	2

and under startup and light load conditions when the preheater section is bypassed.

The water hammer preventive features are more fully described in the section that follows.

As shown on Figure 10.1-1, the valves and the piping downstream thereof are Safety Category I, Quality Group B. The valves and their downstream sections of Category I main feedwater and tempering piping are located in the same Category I valve rooms which house the main steamline isolation valves described in Section 10.3.

10.4.7.3 Water Hammer Prevention Features

Several water hammer prevention features have been designed into the feedwater system. These features are provided to minimize the possibility of various water hammer phenomena in the steam generator preheater, steam generator main feedwater inlet piping and the steam generator upper nozzle feedwater piping. The following discussion is typical for each of the four steam generators and their associated feedwater piping.

10.4.7.3.1 Start-Up, Low Load Conditions

a. Under start-up and low load conditions when NSSS rated flow is less than 15% and temperatures are less than 250° F, feedwater will only be admitted to the upper nozzle of the steam generator by the use of flow through the feedwater bypass tempering line and/or flow through the feedwater preheater bypass line via the feedwater bypass control valve and feedwater preheater bypass valve. The 6-inch diameter upper nozzle is located on the upper shell of the steam generator, below the normal, full power water level. Level control in the steam generator is provided by the feedwater bypass control valve at these conditions.

b. Surface mounted resistance temperature detectors (RTD) are provided on each of the feedwater pipes, leading to and very near the steam generator's upper nozzle to detect during start-up and low load conditions as well as other operating conditions, possible back leakage of steam from the steam generator into the feedwater piping. These RTD's are monitored by the plant process computer and alarmed in the main control room so that actions can be taken to initiate feedwater flow to the upper nozzle before potential feedwater hammer conditions may develop.

10.4.7.3.2 Increasing Load

a. As load increases about 15% of NSSS rated flow and feedwater temperatures rise above 250° F, forward feedwater flushing of the main feedwater piping may be initiated by opening

the feedwater isolation bypass valve. A small controlled flow through the 3-inch feedwater isolation bypass line is provided to flush the main feedwater piping between the isolation valve and the steam generator.

b. Three sets of three RTD's are provided on the main feedwater piping upstream and downstream of the feedwater isolation valve and near the steam generator feedwater nozzle to detect when the feedwater flushing temperature rises above 255° F. Two out of three logic is provided for each set of three RTD's and all three must be satisfied to meet the forward flushing temperature requirements.

c. If flow in the 3-inch feedwater isolation valve bypass line (forward flushing flow) remains above a preset minimum and below a preset maximum and the flushing temperatures remain satisfied, a timed period occurs after which a permissive signal is provided to automatically open the feedwater isolation valves. Automatic opening of a feedwater isolation valve can be blocked by placing its control switch in the main control room in the closed position. This automatic permissive to open occurs after a timed period to allow approximately two volumes of water to be purged from the piping between the feedwater isolation valve and the steam generator main feedwater nozzle. Feedwater flow at the main feedwater flow-element must also be above a preset minimum in order for the feedwater isolation valve to open.

d. After the feedwater isolation valve has opened, the feedwater isolation bypass valve will be manually closed.

e. Prior to opening of the feedwater isolation valve, transfer from the feedwater bypass control valve to the feedwater control valve will occur in order to provide steam generator level control at the higher feedwater flow conditions.

f. If flow to the steam generators remains continuous during a load transient and above a minimum flow rate, feedwater will not be terminated to the main feedwater nozzle even if temperature of the feedwater has dropped below 250° F. Interruption or a reduction in flow below the minimum rate however, will cause the feedwater preheater section of the steam generator to be bypassed.

g. Steam generator low level trips are provided to close all of the feedwater isolation valves, feedwater isolation bypass valves and feedwater preheater bypass valves. Steam generator low pressure trips are provided to close all of the feedwater isolation valves, feedwater isolation bypass valves, feedwater preheater bypass valves and the feedwater bypass tempering valves.

10.4.7.3.3 Split Feedwater Flow

a. Prior to opening of the feedwater isolation valve, the majority of feedwater flow at the lower power level is introduced to the upper nozzle of the steam generator by the preheater bypass pipe.

b. At higher power levels after the feedwater isolation valve has opened, only a small portion of the feedwater flow bypasses the preheater, with the bypass portion contributing to approximately 10% of full feedwater flow at 100% power. This split feedwater flow arrangement provides an approximate 90% of full flow limit to the main feedwater nozzle at higher power levels in order to minimize the potential for tubing vibration in the steam generator. The feedwater flow rate to the steam generator nozzle is monitored and alarmed, if flow rises above approximately 90%, in order for actions to be taken to reduce flow.

c. The preheater bypass valve remains open throughout the start-up and low load conditions, as well as up to and including full power operation.

10.4.7.3.4 Other Upper Nozzle Feedwater Line Uses

Inasmuch as there is water flowing to the upper nozzle of the steam generator during normal operation, and it is the required location for introducing cold fluid into the steam generator, auxiliary feeedwater and chemical feed are connected to the upper nozzle feedwater lines rather than to the main feedwater lines. The chemical feed lines are used to add chemicals directly to the steam generators under low load conditions prior to wet layup. The chemical feed and auxiliary feedwater lines are Safety Category I, Quality Group B out to, and including their isolation valves.

10.4.7.4 Safety Evaluation

The condensate and feedwater systems are not safety-related except as described in Subsection 10.4.7.1.1. If it is necessary to remove a component such as a feedwater heater, pump, or control valve from service, continued operation of the system is possible by use of the multistream arrangement and the provisions for removing from service and bypassing equipment and sections of the system.

An abnormal operational transient analysis of the loss of a feedwater heater string is included in Subsection 15.1.1.

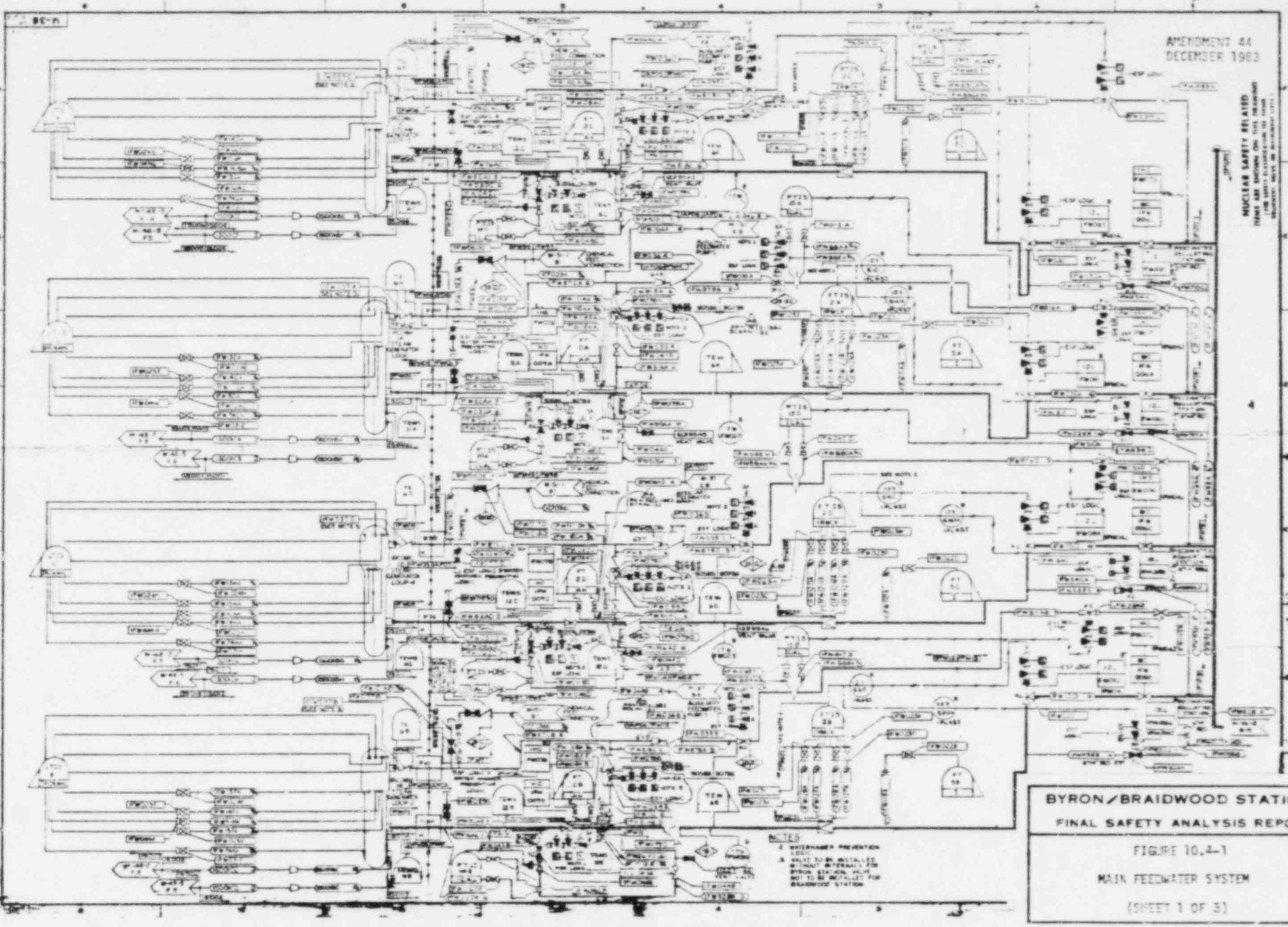

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FIGURE 10.4-1

MAIN FEEDWATER SYSTEM

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errors in the determination of the steady-state power level are made as described in Section 15.0.3.2. The thermal power values used for each transient analyzed are given in Table 15.0-2. In all cases where the 3579 megawatt thermal (MWt) rating is used in an analysis, the resulting transients and consequences are conservative compared to using the 3425 MWt rating.

The values of other pertinent plant parameters utilized in the accident analyses are given in Tables 15.0-3 and 15.0-4.

15.0.3.2 Initial Conditions

For most accidents which are DNB limited, nominal values of initial conditions are assumed (including an appropriate temperature margin to compensate for steam generator tube fouling). The allowances on power, temperature, and pressure are determined on a statistical basis and are included in the limit DNBR, as described in WCAP-8567 (Reference 10). This procedure is known as the "Improved Thermal Design Procedure," and is discussed more fully in Section 4.4.

For accidents which are not DNB limited, or in which the Improved Thermal Design Procedure is not employed the initial conditions are obtained by adding the maximum steady state errors to rated values. The following conservative steady state errors were assumed in the analysis:

a.	Core Power	$\pm$2% allowance for calorimetric error
b.	Average Reactor Coolant System temperature	$\pm$ 4.9°F allowance for controller deadband and measurement error and steam generator fouling penalty
c.	Pressurizer pressure	$\pm$ 30 pounds per square inch (psi) allowance for steady state fluctuations and measurement error

Table 15.0-2 summarizes initial conditions and computer codes used in the accident analysis, and shows which accidents employed a DNB analysis using the improved thermal design procedure.

15.0.3.3 Power Distribution

The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimizes adverse power distribution through the placement of control rods and operating instructions. Power distribution may be characterized by the radial factor ($F_{\Delta H}$) and the total peaking factor (F_Q). The peaking factor limits are given in the technical specifications.