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PLANT NAME: H. B. Robinson Unit #2

ENCLOSURES:

Additional Information For Operation At 2300 MWt Core Power

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Carolina Power & Light Company

May 17, 1974

50-261

File: NG-3514

Serial: NG-74-622

RECEIVED MAY 17 1974 U.S. ATOMIC ENERGY COMMISSION Regulatory Mail Section

Mr. John F. O'Leary, Director
Directorate of Licensing
Office of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. O'Leary:

H. B. ROBINSON UNIT NO. 2
LICENSE DPR-23
ADDITIONAL INFORMATION FOR OPERATION AT 2300 MWt CORE POWER

It has come to our attention that certain information contained in our application of February 1, 1974, for uprating of the H. B. Robinson Unit No. 2 Plant to a core power level of 2300 MWt is erroneous. Specifically, information in Tables 1.4-1 and 3.2.2-1 and on Page 14.3.2-15 pertaining to average and maximum values of thermal output must be revised. Transmitted herewith are three signed originals and forty (40) copies of revised FSAR page changes to correct the above mentioned information.

As required by Commission Regulations, this submittal is signed under oath by a duly authorized officer of the Company.

Yours very truly,

E. E. Utley
Vice-President
Bulk Power Supply

DBW:mvp
Enclosure

Sworn to and subscribed before me this 17th day of May, 1974.

My commission expires: July 4, 1975

4443

Results - Large Break

The following are the results as presented in Reference 27.

The detailed description of the Reactor Coolant System that can be obtained in the multi-control volumes SATAN-V code has been used to analyze important phenomena affecting the blowdown process, such as:

1. Heat transfer from core to the coolant during blowdown

2. Reactor coolant pump characteristics

3. Steam generator heat transfer characteristics

4. Loop resistances and break locations

5. Accumulator performance

A parametric survey study was performed for the Indian Point Unit No. 2 Final Safety Analysis Report [(11)] (Supplements 12 and 13) with the purpose of determining the most conservative combination of the above assumptions as input to the SATAN-V code. Additional parametric studies performed for Turkey Point Unit No. 3 and Indian Point Unit No. 2 are presented in "Emergency Core Cooling Performance"[(3)] submitted to the AEC by Westinghouse. The results presented in this section are based on the conservative assumptions determined in that study.

The results indicated that generally the most conservative assumption for the pump is to assume that the pumps trip at the time of the break and continue to coast down until cavitation conditions were reached. At this time the pump was assumed not to lock but to continue to develop a conservative head. The pump speed is continually calculated as a function of prevailing conditions and the pump characteristics.

The analysis for the 0.5 ft^2 cold leg break using the interim criteria resulted in a peak clad temperature of 2200°F. This result, which is greater than the peak clad temperature calculated for the 3.0 ft.2 cold leg break, is not surprising because a decrease in break size increases the amount of water which is injected during blowdown and which subsequently must be discarded. The

actual transient showed that the clad temperature is nearly equal to the fluid temperature before the end of blowdown. The heatup occurs during the assumed adiabatic period assotiated with reflood of the lower head. On the other hand, this break is below the limit of the break size for which accumulator water loss need be considered because of the maximum amount of water that can be passed through the break and the much lower steam velocities in the system.

Additional studies were performed of a pipe failure in the form of a split with break areas equal to double ended, 0.8 double ended break and 0.6 double ended break. The guillotine rupture simulation was found to be more severe with peak clad temperature more than 100°F higher than the corresponding split cases for the double ended and the 0.8 double ended breaks. The split rupture simulation was found to be more severe for the 0.6 double ended break.

The hot leg break case has not been presented because experience has shown that clad temperatures are low for the following reasons:

1) There is no flow reversal during hot leg break blowdown; the flow just decays smoothly, providing good heat transfer in the core.

2) In a cold leg break analysis, the accumulatory and low head injection flow for the broken loop is assumed to spill to the containment. This assumption is not valid for a hot leg break; thus there is more accumulator water mass and more low head injection flow available for cooling.

3) During the reflood phase of the accident there is no problem with steam binding since the steam generated in the core is vented directly to the containment vía the broken hot leg; heat transfer during reflood for the hot leg break will be better than for the cold leg break. The peak clad temperature will be less than 2000°F.

The analysis of the LOCA was performed at 102% of the core maximum calculated power of 2300 MWt and at a peak linear power of 15.8 kw/ft (equivalent to 15.5 kw/ft at 100% power). This value of peak linear power includes a 5% allowance for nuclear uncertainties.

TABLE 1.4.1

COMPARISON OF DESIGN PARAMETERS

	ROBINSON #2 UPRATING REPORT	TURKEY POINT #3 or #4 FINAL REPORT	INDIAN POINT #2 FINAL REPORT	GINNA FINAL REPORT	REFERENCE LINE NO.
HYDRAULIC AND THERMAL DESIGN PARAMETERS					
Total Primary Heat Output, MWt	2308	2200	2758	1300	1
Total Core Heat Output, Btu/hr	7850×10^6	7479×10^6	9413×10^6	4437×10^6	2
Heat Generated in Fuel, %	97.4	97.4	97.4	97.4	3
Maximum Thermal Overpower	12%	12%	12%	12%	4
System Pressure, Nominal, psia	2250	2250	2250	2250	5
System Pressure, Minimum Steady State, psia	2220	2220	2220	2220	6
Hot Channel Factors					
Heat Flux, F_q	2.65	3.23	3.23	3.38	7
Enthalpy Rise, $F_{\Delta H}$	1.55	1.77	1.77	1.77	8
DNB Ratio at Nominal Conditions	2.02	1.81	2.00	2.15	9
Minimum DNBR for Design Transients	1.30	1.30	1.30	1.30	10
Coolant Flow					
Total Flow Rate, lb/hr	101.5×10^6	101.5×10^6	136.3×10^6	67.3×10^6	11
Effective Flow Rate for Heat Transfer, lb/hr	97.0×10^6	97.0×10^6	$130. \times 10^6$	64.3×10^6	12
Effective Flow Area for Heat Transfer, ft^2	41.8	41.8	51.4	27.0	13
Average Velocity Along Fuel Rods, ft/sec	14.3	14.3	15.4	14.7	14
Average Mass Velocity, lb/hr-ft^2	2.32×10^6	2.32×10^6	2.53×10^6	2.38×10^6	15
Coolant Temperatures, °F					
Nominal Inlet	546.1	546.2	543	551.9	16
Maximum Inlet Due to Instrumentation Error and Deadband, °F	550.1	550.2	547	555.9	17
Average Rise in Vessel, °F	58.5	55.9	53.0	49.5	18
Average Rise in Core	61.0	58.3	55.5	52	19
Average in Core	577.8	575.4	571.0	578.0	20
Average in Vessel	575.4	574.2	569.5	577.0	21
Nominal Outlet of Hot Channel	642	642	633.5	634.0	22
Average Film Coefficient, Btu/hr-ft^2-F	5560	5400	5790	5590	23
Average Film Temperature Difference, °F	32.6	31.8	30.3	26.9	24
Heat Transfer at 100% Power					
Active Heat Transfer Surface Area, ft^2	42,460	42,460	52,200	28,715	25
Average Heat Flux, Btu/hr-ft^2	181,000	171,600	175,600	150,500	26
Maximum Heat Flux, Btu/hr-ft^2	488,100	554,200	567,300	508,700	27
Average Thermal Output, kw/ft	5.83	5.5	5.7	4.88	28
Maximum Thermal Output, kw/ft	15.5	17.9	18.4	16.5	29

TABLE 1.4.1 (Cont'd)

	ROBINSON #2 UPRATING REPORT	TURKEY POINT #3 or #4 FINAL REPORT	INDIAN POINT #2 FINAL REPORT	GINNA FINAL REPORT	REFERENCE LINE NO.
Maximum Clad Surface Temperature at Nominal Pressure, °F	657	657	657	657	30
Fuel Central Temperature, °F					
Maximum at 100% Power	3800	4030	4090	3880	31
Maximum at Overpower	4100	4300	4380	4100	32
Thermal Output, kw/ft at Maximum Overpower	17.7	20.0	20.6	18.5	33
CORE MECHANICAL DESIGN PARAMETERS					
Fuel Assemblies					
Design	RCC Canless 15 x 15	RCC Canless 15 x 15	RCC Canless 15 x 15	RCC Canless 14 x 14	34
Rod Pitch, in.	0.563	0.563	0.563	0.556	35
Overall Dimensions, In.	8.426 x 8.426	8.426 x 8.426	8.426 x 8.426	7.763 x 7.763	36
Fuel Weight (as UO_2), pounds	175,400	176,200	216,000	120,872	37
Total Weight, pounds	225,400	226,200	276,000	152,895	38
Number of Grids per Assembly	7	7	9	9	39
Fuel Rods					
Number	32,028	32,028	39,372	21,659	40
Outside Diameter, in.	0.422	0.422	0.422	0.422	41
Diametral Gap, in.	0.0075	0.0065	0.0065	0.0065	42
Clad Thickness, in.	0.0243	0.0243	0.0243	0.0243	43
Clad Material	Zircaloy	Zircaloy	Zircaloy	Zircaloy	44
Fuel Pellets					
Material	UO_2 Sintered	UO_2 Sintered	UO_2 Sintered	UO_2 Sintered	45
Density (% of Theoretical)	94-95	94-92-91	94-92-91	94-93	46
Diameter, in.	0.3659	0.3669	0.3669	0.3669	47
Length, in.	0.6000	0.6000	0.6000	0.6000	48
Rod Cluster Control Assemblies					
Neutron Absorber	5% Cd-15% In-80% Ag.	5% Cd-15% In-80% Ag.	5% Cd-15% In-80% Ag	5% Cd-15% In-80% Ag	49
Cladding Material	Type 304 SS-Cold Worked	Type 304 SS-Cold Worked	Type 304 SS-Cold Worked	Type 304 SS-Cold Worked	50
Clad Thickness, in.	0.019	0.019	0.019	0.019	51
Number of Clusters	53	53	53	29	52
Number of Control Rods per Cluster	20	20	20	16	53
Core Structure					
Core Barrel I.D./O.D., in.	133.875/137.875		148.0/152.5	109.0/112.5	54
Thermal Shield I.D./O.D., in.	142.625/148.0		158.5/164.0	115.3/122.5	55
FINAL NUCLEAR DESIGN DATA					
Structural Characteristics					
Fuel Weight (As UO_2), lbs.	175,400	176,200	216,000	120,130	56
Clad Weight, lbs.	36,300	36,300	44,600	22,440	57
Core Diameter, in. (Equivalent)	119.7	119.5	132.5	96.5	58

TABLE 3.2.2-1

THERMAL AND HYDRAULIC DESIGN PARAMETERS

Total Primary Heat Output, MWt	2308
Total Reactor Coolant Pump Heat Output, MWt	8
Total Core Heat Output, MWt	2300
Total Core Heat Output, Btu/hr	7850×10^6
Heat Generated in Fuel, %	97.4
Maximum Thermal Overpower, %	12
Nominal System Pressure, psia	2250
Hot Channel Factors	
Heat Flux	
Nuclear, F_q^N	2.57
Engineering, F_q^E	1.03
Total	2.65
Enthalpy Rise	
Nuclear, F_H^N	1.55
Coolant Flow	
Total Flow Rate, lbs/hr	101.5×10^6
Average Velocity Along Fuel Rods, ft/sec	14.3
Average Mass Velocity, lb/hr-ft^2	2.32×10^6
Coolant Temperature, °F	
Nominal Inlet	546.1
Average Rise in Vessel	58.5
Average Rise in Core	61.0
Average in Core	577.8
Average in Vessel	575.4
Nominal Outlet of Hot Channel	642.
Heat Transfer	
Active Heat Transfer Surface Area, ft^2	42,460
Average Heat Flux, Btu/hr-ft^2	181,000
Maximum Heat Flux, Btu/hr-ft^2	488,100
Maximum Thermal Output, kw/ft	15.5
Maximum Clad Surface Temperature at Nominal Pressure, °F	657
Maximum Average Clad Temperature at Rated Power, °F	715
Fuel Central Temperatures, °F	
Maximum at 100% Power	3,800
Maximum at 112% Power	4,100
DNB Ratio	
Minimum DNB Ratio at nominal operating conditions	2.02
Pressure Drop, psi	
Across Core	26
Across Vessel, including nozzles	46

TABLE 3.2.2-2

ENGINEERING HOT CHANNEL FACTORS

F_q^E	Pellet Diameter, Density Enrichment, and Eccentricity Rod Diameter, (Pitch and Bowing)	1.03
$F_{\Delta H}^E$	Pellet Diameter, Density, Enrichment Rod Diameter, Pitch and Bowing	1.08
	Inlet Flow Maldistribution	1.01
	Flow Redistribution	1.03
	Flow Mixing	0.90*
	Resulting $F_{\Delta H}^E$	1.01

* To point of Minimum DNB ratio

CAROLINA POWER & LIGHT COMPANY

H. B. ROBINSON UNIT 2

DOCKET NO. 50-261

INSTRUCTION SHEET

This amendment contains information in support of CP&L's request for an Amendment to the Operating License to allow operation at 2300 MWt core power. The revised pages contain the results of the analyses done in support of 2300 MWt operation as well as other information important to the review.

Each revised page bears the date May, 1974, in the upper right hand corner. Vertical bars have been used in the margins of the revised pages to indicate the location of the revision on the page.

The following page removals and insertions should be made to incorporate these page changes into the FSAR.

REMOVE (Existing Pages)	INSERT (Amendment Pages)
1.4-4/1.4-5	1.4-4/1.4-5
Tables 3.2.2-1/3.2.2-2	Tables 3.2.2-1/3.2.2-2
14.3.2-14/14.3.2-15	14.3.2-14/14.3.2-15