VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

July 23, 1979

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. Albert Schwencer, Chief
Operating Reactors Branch No. 1
Division of Reactor Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Serial No. 567
FR/MLB/RWC
LQA/DWS
Docket No. 50-281

License No. DPR-37

Dear Mr. Denton:

RELOAD APPLICATION FOR CYCLE 5
SURRY NUCLEAR POWER STATION UNIT NO. 2

In order to facilitate NRC review of the Cycle 5 reload core for Surry Unit No. 2, the reload application will be submitted in two parts. Part 1, which is attached, provides a discussion of the design of the reload core. Significant areas that are addressed include the continued use of one 17 x 17 Demonstration Assembly and the fresh feed fuel region which was refabricated. As you know, based on discussions between our Mr. M. L. Bowling and your Mr. D. Neighbors that 62 of the fresh feed assemblies and 41 of the burnable poison assemblies planned for use in Cycle 5 were chemically contaminated and are currently being refabricated by Westinghouse at the Columbia, South Carolina facility. We expect these assemblies to be returned to Surry in essentially their original condition during late August and September 1979. As a result of our detailed investigation and plant inspection and the information we have obtained from the investigating authorities, we have concluded, as of this date, that only the fuel assemblies and fuel components identified above were affected. A related inspection was conducted by NRC Region II and is documented in their letter of June 22, 1979 (Inspection Reports Nos. 50-280/79-25 and 50-281/79-37).

This part of this application has been reviewed by both the Station Nuclear Safety and Operating Committee and the System Nuclear Safety and Operating Committee. It has been determined that no unreviewed safety questions as defined in 10 CFR 50.59 will exist from the use of either the fresh feed fuel region or the 17 x 17 Demonstration Assembly in Cycle 5.

The second part of the application will be submitted in approximately 30 days, and will address the Cycle 5 reload design impact on transients delineated in Chapter 14 of the Final Safety Analysis Report. Any necessary Technical Specification changes will be identified and discussed at that time.

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Should you have questions or comments, we would be most happy to meet with you at your earliest convenience.

Very truly yours,

W. N. Thomas
Vice President
Fuel Resources

Attachment

cc: Mr. James P. O'Reilly, Director
Office Inspection and Enforcement
Region II

ATTACHMENT 1

1.0 Introduction

Part 1 of this Reload Safety Evaluation (RSE) report presents a description of the design for the Cycle 5 reload core of Unit 2 of the Surry Nuclear Power Station*.

Unit 2 has completed its fourth cycle of operation and is currently undergoing steam generator replacement. The unit is projected to be refueled during September of 1979. During the refueling, 64 fuel assemblies will be replaced with fresh assemblies. The loading pattern also includes 93 previously irradiated fuel assemblies including a single 17 x 17 Demonstration Assembly and 56 fresh and depleted burnable poison assemblies. The demonstration assembly is being reinserted as part of a joint Vepco/Westinghouse/DOE program to demonstrate the feasibility of extended discharge burnups. It is planned that Unit 2 will begin Cycle 5 operation in early Fall 1979 and that the cycle will extend to Spring 1981 for a nominal 18-month cycle, producing approximately 13,500 MWD/MTU of energy. Operation beyond 13,500 MWD/MTU is possible in a power coastdown mode.

All design analyses performed for the Cycle 5 reload core are to be based on adherence to the following conditions unless otherwise stated in this document:

1) Plant operating limits delineated in the Technical Specifications, to include those pending changes to the specifications provided in Reference 1.
2) Cycle 5 operation not in excess of 14,500 MWD/MTU.
3) The design methodology documented in References 2 and 3.

*Part 2 of the RSE will address the thermal-hydraulic and core physics considerations as they apply to the safety analysis evaluation, any necessary Technical Specifications changes and the startup physics testing program.

2.0 Design

2.1 Basic Design Parameters

The basic design parameters for Cycle 5 are a core average power of 2441 Mwt, a system pressure of 2250 psia, a core average temperature of 574.4°F, a core linear power density of 6.2 kw/ft, and a minimum total core flow of 265,500 GPM.(2) These parameters are consistent with the bases for the Steam Generator Replacement Program as documented in Reference 4 and approved in Reference 5.

2.2 Design Core Loading Pattern

The fuel assembly configuration for the Cycle 5 reload core is shown in Figure 1, and significant fuel assembly design parameters are denoted in Table 1. The core loading will contain 688 fresh borosilicate burnable poison rods and 48 depleted burnable poison rods. The burnable poison rod locations and distributions are shown in Figure 2. Also shown in Figure 2 are the locations of the primary sources, secondary sources and control rods which will be incorporated into the Cycle 5 core.

2.3 Nuclear Design

Representative radial power distributions have been calculated with the model documented in Reference 6 and are provided in Figures 3-6. Assembly-wise beginning-of-cycle burnup distributions are provided in Figure 1 and conservative region average end of cycle burnups are provided in Table 1. Burnup calculations were performed with the model documented in Reference 7. These power distributions are very similar to those power distributions experienced in previous Surry operating cycles.

Design predictions to support the startup physics testing program are identified in Reference 8 and will be calculated with the models documented in References 6, 7, and 9.

2.4 Design Changes

The unit's steam generators are being replaced as documented in References 4 and 5. From a reload core design standpoint, the new steam generators are identical to those being replaced.

2.5 Fuel Design

The mechanical design of the fresh Region 7 fuel assemblies is identical to Region 6, which is described in Reference 10. Integrity and performance capability of the Region 4, 5, and 6 fuel to be reinserted in Cycle 5 is considered very good based on the coolant activity levels and core performance observed during Cycle 4 operation[11] and on routine visual inspection of this fuel after removal from the core. Extensive experience with Zircaloy-clad fuel of the type being used in this reload core has been obtained, and this experience is described in Reference 12. This report is updated periodically.

The Region 7 fuel has been designed with the fuel performance model documented in Reference 13. The internal pressure of the lead rod in the reactor is limited to a value below that which would cause the diametral gap between the fuel and clad to increase due to outward cladding creep during steady-state operation. The DNB propagation criteria delineated in Reference 14 are satisfied.

Clad flattening will not occur during Cycle 5. Clad flattening time is predicted to be greater than 30,000 EFPH for all 15 x 15 fuel regions being irradiated using the model documented in Reference 15. The most limiting fuel region, 5A, currently has an accumulated core residence time of 16,300 EFPH. Therefore, Region 5A has an allowed residence time of 13,700 EFPH in Cycle 5. Cycle 5 operation will not exceed 10,300 EFPH.

2.5.1 Extended Burnup Demonstration Program

The impact of additional irradiation of the 17 x 17 Demonstration Assembly during Cycle 5 was specifically addressed, and it was determined that all 17 x 17 fuel performance criteria were satisfied. WCAP-8362[16], as supplemented by Reference 17, concludes that the presence of a 17 x 17 Demonstration Assembly does not affect reactor performance adversely relative to an all 15 x 15 assembly core.

The limiting criteria of clad flattening and internal rod pressure demonstrate considerable margin to the design criteria. The clad flattening time is predicted to be greater than 40,000 EFPH for the 17 x 17 fuel design using the approved Westinghouse Evaluation Model.[15] The demonstration assembly currently has a cumulative core residence time of approximately 22,600 EFPH. Therefore, an additional 17,400 EFPH of allowable exposure exists, which is well above the Cycle 5 maximum allowable value of 10,300 EFPH. The internal pressure of the lead rod in the demonstration assembly is limited to a value below that which would cause the diameteral gap between the fuel and clad to increase due to outward cladding creep during steady state operation. The DNB propagation criteria delineated in Reference 14 are satisfied. The fuel performance model used for these evaluations is documented in Reference 13.

The 17 x 17 Demonstration Assembly (which is designed to allow for the removal of individual fuel rods) will undergo a detailed examination before refueling. It is planned that four of the fuel rods in the assembly will be replaced by four previously irradiated fuel rods. The replacement of these rods does not alter the conclusion that a 17 x 17 Demonstration Assembly will not adversely affect reactor performance relative to an all 15 x 15 core.[18] Considerable experience with removing 17 x 17 fuel rods from demonstration assemblies for either inspection or replacement has been obtained.

2.5.2 Fuel Assembly and Burnable Poison Assembly Refabrication

Most of the fresh fuel assemblies (62 of 64 assemblies) and 41 of the burnable poison assemblies designated for initial irradiation in this cycle were contaminated with an unknown amount of industrial (technical) grade caustic soda (NaOH). Chemical analysis of samples obtained from each of the fuel assemblies confirmed the identity of the contaminant. The constituents of the contaminant that were of potential metallurgical concern to the fuel and burnable poison assemblies were sodium and chlorine (sodium to chlorine ratio averaged approximately 70). Detailed inspection at the storage site of the contaminated assemblies indicated no metallurgical impact, as expected, since the temperature and stress conditions necessary for initiation of metallurgical damage were not present in the storage location. Additional evaluations carried out in both laboratory and manufacturing environments confirmed that no metallurgical impact had occurred.

To ensure adequate decontamination of the fuel and burnable poison assemblies, all of the fuel (i.e., 64 assemblies including 2 assemblies not suspected of being contaminated) and burnable poison assemblies were returned to the fuel supplier (Westinghouse). In order to develop a decontamination procedure, the fuel supplier has subjected both fully and partially disassembled fuel assembly components to various cleaning alternatives. Based on the results, a decision was made to disassemble all of the fuel and burnable poison assemblies and reuse, after cleaning, all components except the top six grids, thimble guide tubes, and various screws/bolts. New grids (except the bottom grid) and thimble guide tubes will be used in the refabricated assembly. The production decontamination process consists of wiping fuel and burnable poison rods with tap water and again with acetone. Fuel assembly top and bottom nozzles, bottom grids, thimble plugs and burnable poison baseplates are to be bathed ultrasonically in a weak nitric acid solution followed by a tap water rinse. All cleaned components will undergo a detailed visual inspection before being accepted for incorporation into the refabricated fuel and burnable poison assemblies, which will then be subjected to the same quality control, inspection and pre-shipment cleaning required during the normal manufacturing process.

Acceptance criteria for allowable concentrations of sodium and chlorine were developed based on a conservative interpretation of currently available metallurgical data and then arbitrarily reduced by a factor of 20 for additional conservatism. A decontamination (i.e., cleaning) process was then developed to meet these acceptance criteria. To demonstrate that the acceptance criteria could be met, a laboratory program to qualify the integrity of the fuel rods and the decontamination process was conducted. This qualification consisted of autoclave testing, detailed visual inspection, and chemical analysis of pre-characterized quantities of the contaminant after the production decontamination had been completed. The autoclave testing was performed on cut sections of five fuel rods which included the heat treated and welded areas of the fuel rod. These five fuel rods will be replaced with new fuel rods. Leak testing of each fuel rod is being performed as part of the refabrication process. Based on the results from this qualification program, it was concluded that the production decontamination process will meet the conservative acceptance criteria.

No design or manufacturing tolerance criteria will be changed for the refabricated fuel and burnable poison assemblies. These refabricated assemblies, when inserted into Cycle 5 will be identical to the fuel and burnable poison assemblies originally supplied, except for superficial scratches and blemishes which routinely occur when the assemblies are disassembled and reassembled. Consequently, there will be no impact on either performance or safety from the use of this fuel.

3.0 References

1. Letter from C. M. Stallings (Vepco) to H. R. Denton (NRC), Serial No. 388, May 31, 1979.

2. Final Safety Analysis Report, Surry Power Station, Units 1 and 2.

3. Bordelon, F. M. et.al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272, March 1978.

4. Letter from C. M. Stallings (Vepco) to E. G. Case (NRC) dated August 17, 1977, Serial No. 351 as revised.

5. Letter from A. Schwencer (NRC) to C. M. Stallings (Vepco) dated December 15, 1978.

6. Letter from W. N. Thomas (Vepco) to B. C. Rusche (NRC), Serial No. 166, July 26, 1976, (VEP-FRD-19, "The PDQ07 Discrete Model, Virginia Electric and Power Company").

7. Letter from W. N. Thomas (Vepco) to B. C. Rusche (NRC), Serial No. 011, January 25, 1977, (VEP-FRD-20, "The PDQ07 One Zone Model, Virginia Electric and Power Company").

8. Letter from C. M. Stallings (Vepco) to E. G. Case (NRC), Serial No. 108, March 15, 1978.

9. Letter from W. N. Thomas (Vepco) to H. R. Denton (NRC), Serial No. 017, January 9, 1979, (VEP-FRD-24, "The Vepco FLAME Model, Virginia Electric and Power Company").

10. Letter from C. M. Stallings (Vepco) to E. G. Case (NRC), Serial No. 403, September 9, 1977.

11. Letter from W. N. Thomas (Vepco) to E. G. Case (NRC), Serial No. 303, April 27, 1979, (VEP-FRD-32, "Surry Unit 2, Cycle 4 Core Performance Report").

12. O'Hara, T. L., and Iorii, J. A., "Operational Experience with Westinghouse Cores," WCAP-8183, Revision 7, March 1978.

13. J. V. Miller, "Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations," WCAP-8720, November 1976.

14. "Risher, D. H. et.al., "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis," WCAP-8964, June, 1977.

15. George, R. A., et.al., "Revised Clad Flattening Model," WCAP-8377 (Proprietary) and WCAP-8381 (Non-proprietary), July 1974.

16. WCAP-8362, "Irradiation of 17 x 17 Demonstration Assemblies in Surry Units No. 1 and 2, Cycle 2," July 1974.

17. Letter from C. M. Stallings (Vepco) to B. C. Rusche (NRC), Serial No. 763, October 30, 1975.

18. Letter from C. M. Stallings (Vepco) to B. C. Rusche (NRCO, Serial No. 936, March 11, 1976.

TABLE 1

SURRY UNIT 2-CYCLE 5

FUEL ASSEMBLY DESIGN PARAMETERS

Region	4B	5A	6A	6B	7A	7B
Enrichment (w/o U235)	3.10	3.11	2.91	3.20	3.1*	3.4*
Density (% Theoretical)	94.6	94.5	94.5	94.7	95*	95*
Number of Assemblies	1	24	20	48	12	52
Approximate Burnup at Beginning of Cycle (MWD/MTU)	28200	22000	15800	11900	0	0
Estimate of Burnup at End-of-Cycle 5 (MWD/MTU)	42100	33800	27800	26600	17500	15800
MTU per Region	0.458	10.97	9.14	22.0	5.51	23.86

*Nominal

Figure 1

SURRY UNIT 2, CYCLE 5
FUEL LOADING AND BOC FUEL ASSEMBLY BURNUP
(13678 MWD/MTU EOC4)

	08	09	10	11	12	13	14	15
H	4B L08 28171	7A 12 FRESH 0	6A L12 15415	6B 12* J15 8948	6A H09 15775	6B N10 15457	6B R09 8948	7B FRESH 0
J	7A 12 FRESH 0	6A M11 15415	7B 16 FRESH 0	5A L09 21295	7B 16 FRESH 0	6B L13 13848	7B 12 FRESH 0	6A J10 16247
K	6A L12 15415	7B 16 FRESH 0	6B N12 9729	5A K11 22811	6B P09 14395	7B 16 FRESH 0	7B FRESH 0	
L	6B 12* J15 8948	5A J11 21295	5A L10 22811	6B M13 9729	7A 12 FRESH 0	7B 8 FRESH 0	5A J12 22024	
M	6A J08 15775	7B 16 FRESH 0	6B J14 14395	7A 12 FRESH 0	6B K13 15457	6B L14 9191		
N	6B SS N10 15457	6B N11 13848	7B 16 FRESH 0	7B 8 FRESH 0	6B P11 9191			
P	6B R09 8948	7B 12 FRESH 0	7B FRESH 0	5A M09 22024				
R	7B FRESH 0	6A K09 16247						

Batch No.	No. of Fuel Assemblies	Initial Enrichment w/o U235
4B	1	3.1
5A	24	3.1
6A	20	2.9
6B	48	3.2
7A	12	3.1
7B	52	3.4

LEGEND

BATCH NO	XX	YY	No. of Fresh Burnable Poison Rods/Secondary Source Location
		ZZ	Previous Cycle location (if applicable)
		LL	Beginning of Cycle burnup in MWD/MTU

Total No. Fresh BP Rods = 688
Total No. Depleted BP Rods = 48
*Depleted BP

FIGURE 2
SOURCE, CONTROL ROD AND BURNABLE POISON ROD LOCATIONS AND DISTRIBUTION
SURRY UNIT 2-CYCLE 5

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2						R	PS 12	R	12	R					
3					8	16	R	SS	R	16	8				
4				R	12	R	16		16	R	12	R			
5			8	12	R			12*			R	12	8		
6		R	16	R		R	16	R	16	R		R	16	R	
7		12	R	16		16	R	12	R	16		16	R	12	
8		R			12*	R	12		12	R	12*			R	
9		12	R	16		16	R	12	R	16		16	R	12	
10		R	16	R	FSS	R	16	R	16	R		R	16	R	
11			8	12	R			12*			R	12	8		
12				R	12	R	16		16	R	12	R			
13					8	16	R	SS	R	16	8				
14						R	12	R	PS 12	R					
15															

North

R - Full Length Control Rod
X - Number of Fresh Burnable Poison
X*- Depleted Burnable Poison Rod Assemblies
PS- Primary Source Location
SS- Secondary Source Location
FSS- Fresh Secondary Source Location

Figure 3

SURRY 2, CYCLE 5, ARO MAP, 0 MWD/MTU (HFP)

0.889 0.946							
1.161 1.308	1.094 1.142						
1.110 1.157	1.201 1.329	1.234 1.327					
1.189 1.280	1.029 1.097	1.017 1.058	1.223 1.321				
1.090 1.144	1.192 1.326	1.163 1.236	1.137 1.283	0.893 1.081			
1.191 1.269	1.197 1.283	1.150 1.278	1.008 1.261	0.542 0.983			
1.135 1.206	1.062 1.244	0.946 1.290	0.408 0.851				
0.757 1.083	0.422 0.809						

x----------Assembly Relative Power Density

y----------Peak Pin Power

Figure 4

SURRY 2, CYCLE 5, D BANK IN MAP, 0 MWD/MTU (HFP)

0.933 0.983							
1.191 1.348	1.069 1.143						
1.121 1.179	1.085 1.233	0.656 0.989					
1.240 1.324	1.031 1.104	0.968 1.129	1.305 1.367				
1.148 1.206	1.269 1.418	1.286 1.402	1.320 1.490	1.081 1.293			
1.093 1.243	1.204 1.355	1.277 1.427	1.195 1.470	0.668 1.191			
0.539 0.882	0.915 1.242	1.013 1.371	0.477 0.977				
0.500 0.668	0.329 0.549						

x----------------Assembly Relative Power Density

y----------------Peak Pin Power

Figure 5

SURRY 2, CYCLE 5, ARO MAP, 14000 MWD/MTU (HFP)

0.977 0.987							
1.262 1.308	1.100 1.130						
1.076 1.121	1.275 1.332	1.125 1.150					
1.097 1.156	0.993 1.042	0.961 0.998	1.112 1.144				
1.023 1.054	1.246 1.302	1.108 1.172	1.201 1.271	0.924 1.072			
1.055 1.099	1.099 1.173	1.239 1.315	1.092 1.271	0.611 1.021			
1.015 1.060	1.066 1.200	0.923 1.184	0.471 0.855				
0.733 0.980	0.457 0.761						

x------------Assembly Relative Power Density

y------------Peak Pin Power

Figure 6

SURRY 2, CYCLE 5, D BANK IN MAP, 14000 MWD/MTU (HFP)

1.016 1.024							
1.287 1.332	1.070 1.142						
1.077 1.124	1.151 1.266	0.584 0.885					
1.135 1.177	0.989 1.036	0.909 1.052	1.179 1.246				
1.074 1.115	1.324 1.397	1.217 1.322	1.376 1.455	1.098 1.260			
0.971 1.085	1.110 1.242	1.370 1.468	1.273 1.472	0.737 1.212			
0.474 0.784	0.925 1.192	0.984 1.266	0.545 0.972				
0.498 0.623	0.364 0.572						

x------------Assembly Relative Power Density

y------------Peak Pin Power