NFE TECHNICAL REPORT NO. 289
RELOAD SAFETY EVALUATION FOR
SURRY 1 CYCLE 7 REDESIGNED CORE
(PART 2)

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TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

1.0 INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

THIS REPORT PRESENTS AN EVALUATION FOR SURRY POWER STATION UNIT 1, CYCLE 7, WHICH DEMONSTRATES THAT THE REDESIGNED CYCLE 7 CORE WILL NOT ADVERSELY AFFECT THE SAFETY OF THE PLANT. THIS EVALUATION WAS ACCOMPLISHED UTILIZING THE METHODOLOGY DESCRIBED IN WCAP-9272, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY" (REF. 1).

BASED UPON THE ABOVE REFERENCED METHODOLOGY, ONLY THOSE INCIDENTS ANALYZED AND REPORTED IN THE FSAR (REF. 2) WHICH COULD POTENTIALLY BE AFFECTED BY THE FUEL RELOAD HAVE BEEN REVIEWED FOR THE CYCLE 7 DESIGN DESCRIBED HEREIN. NO NEW TRANSIENT ANALYSES WERE REQUIRED FOR THE CYCLE 7 DESIGN. THE JUSTIFICATION FOR THE APPLICABILITY OF PREVIOUS RESULTS IS PROVIDED.

1.2 GENERAL DESCRIPTION

THE SURRY 1 REACTOR CORE IS COMPRISED OF 157 FUEL ASSEMBLIES ARRANGED IN THE CONFIGURATION SHOWN IN FIGURE 1. DURING THE CYCLE 6/7 REFUELING, 94 FUEL ASSEMBLIES WERE REPLACED WITH 64 FRESH REGION 9 ASSEMBLIES AND WITH 30 ASSEMBLIES IRRADIATED IN EARLIER CYCLES. THE PATTERN FOR CYCLE 7 IS SHOWN IN FIGURE 1. A SUMMARY OF THE CYCLE 7 FUEL INVENTORY IS GIVEN IN TABLE 1.

NOMINAL CORE DESIGN PARAMETERS UTILIZED FOR CYCLE 7 ARE AS FOLLOWS:

CORE POWER (MWT)	2441
SYSTEM PRESSURE (PSIA)	2250
VESSEL AVERAGE TEMPERATURE (-F)	574.4
THERMAL DESIGN FLOW (GPM)	265,500
AVERAGE LINEAR POWER DENSITY (KW/FT) (BASED ON HOT, DENSIFIED CORE AVERAGE STACK HEIGHT OF 143.6 INCHES)	6.2

1.3 CONCLUSIONS

FROM THE EVALUATION PRESENTED IN THIS REPORT, IT IS CONCLUDED THAT THE CYCLE 7 DESIGN DOES NOT RESULT IN THE PREVIOUSLY ACCEPTABLE SAFETY LIMITS BEING EXCEEDED FOR ANY INCIDENT AND CONSEQUENTLY NO UNREVIEWED SAFETY QUESTIONS EXIST AS A RESULT OF THIS RELOAD. THE CONCLUSIONS ARE BASED ON THE FOLLOWING:

1. AN ACTUAL CYCLE 6 BURNUP OF 16491 MWD/MTU
2. CYCLE 7 BURNUP WILL NOT EXCEED 13000 MWD/MTU (NOMINAL END OF REACTIVITY PLUS APPROXIMATELY 1500 MWD/MTU OF COASTDOWN)
3. THERE IS ADHERENCE TO PLANT OPERATING LIMITATIONS AS GIVEN IN THE TECHNICAL SPECIFICATIONS AND THE AMENDMENT THERETO PROPOSED IN SECTION 4

2.0 REACTOR DESIGN

2.1 MECHANICAL DESIGN

THE MECHANICAL DESIGN OF THE REGION 9 FUEL ASSEMBLIES IS THE SAME AS THE REGION 8 ASSEMBLIES. TABLE 1 COMPARES PERTINENT DESIGN PARAMETERS OF THE VARIOUS FUEL REGIONS. THE REGION 9 FUEL HAS BEEN DESIGNED ACCORDING TO THE FUEL PERFORMANCE MODEL IN REFERENCE 3. THE FUEL IS DESIGNED AND OPERATED SO THAT CLAD FLATTENING WILL NOT OCCUR, AS PREDICTED BY THE WESTINGHOUSE MODEL (REF. 4). FOR ALL FUEL REGIONS, THE FUEL ROD INTERNAL PRESSURE DESIGN BASIS, WHICH IS ACCEPTABLE AS SHOWN IN REFERENCE 5, IS SATISFIED.

WESTINGHOUSE HAS HAD CONSIDERABLE EXPERIENCE WITH ZIRCALOY CLAD FUEL. THIS EXPERIENCE IS EXTENSIVELY DESCRIBED IN WCAP-8183, "OPERATIONAL EXPERIENCE WITH WESTINGHOUSE CORES" (REF. 6). THIS REPORT IS UPDATED ANNUALLY.

2.2 NUCLEAR DESIGN

THE CYCLE 7 CORE LOADING HAS A LOCA FQ LIMIT OF 2.18 UNDER NORMAL OPERATING CONDITIONS. THE MAXIMUM ANALYTICALLY PREDICTED FQ FOR CYCLE 7 IS 2.15; FQ IS LESS THAN THE LIMIT AT ALL CORE ELEVATIONS FOR THIS CYCLE. THEREFORE, FREQUENT AXIAL POWER DISTRIBUTION MONITORING IS NOT REQUIRED.

TABLE 2 PROVIDES A SUMMARY OF CHANGES IN THE CYCLE 7 KINETICS CHARACTERISTICS COMPARED WITH THE CURRENT LIMITS BASED ON PREVIOUSLY SUBMITTED ACCIDENT ANALYSES. AS SHOWN IN THE TABLE, ONLY ONE OF THE CYCLE 7 PARAMETERS, THE MOST NEGATIVE DOPPLER TEMPERATURE COEFFICIENT, FALLS OUTSIDE ITS CURRENT LIMIT. THIS PARAMETER IS EVALUATED IN SECTION 3.0.

TABLE 3 PROVIDES THE CONTROL ROD WORTHS AND REQUIREMENTS AT THE MOST LIMITING CONDITIONS DURING THE CYCLE. THE REQUIRED SHUTDOWN MARGIN IS BASED ON PREVIOUSLY SUBMITTED ACCIDENT ANALYSES (REF. 2). THE AVAILABLE SHUTDOWN MARGIN EXCEEDS THE MINIMUM REQUIRED.

WHILE ALL OTHER DESIGN CONSTRAINTS HAVE BEEN ASSESSED AND ARE MET FOR THE CURRENT ROD INSERTION LIMITS, THE PREDICTED RADIAL POWER PEAKING FOR CYCLE 7 IS IN EXCESS OF THE CURRENT DESIGN LIMIT AT HOT FULL POWER (ONLY FOR BURNUPS LESS THAN 1000 MWD/MTU) AND AT HOT ZERO POWER (MAJORITY OF CYCLE LIFE) FOR THE CONTROL ROD INSERTION LIMITS OF THE CURRENTLY APPROVED TECHNICAL SPECIFICATIONS. AS A RESULT, A CHANGE TO THE TECHNICAL SPECIFICATIONS IS PROPOSED WHICH RAISES THE ROD INSERTION LIMITS. AT THE REVISED INSERTION LIMITS, PROVIDED IN FIGURE 2, THE RADIAL PEAKING FACTORS ARE WITHIN THE APPROPRIATE CURRENT DESIGN LIMITS. (HOWEVER, IT SHOULD BE NOTED THAT THE CURRENT ROD INSERTION LIMITS WOULD BE ACCEPTABLE AFTER ATTAINING A BURNUP OF 1000 MWD/MTU AND IF THE RADIAL POWER PEAKING FACTOR DESIGN LIMIT WERE BASED ON

A 0.3 PART POWER MULTIPLIER, I.E.,

$$FDH = 1.55(1+0.3(1-P))$$

$$P = \text{FRACTION OF RATED POWER}$$

INSTEAD OF THE 0.2 VALUE WHICH IS CURRENTLY APPROVED.)

THE LOADING CONTAINS A TOTAL OF 608 FRESH BURNABLE POISON RODS LOCATED IN 52 OF THE REGION 9 FUEL ASSEMBLIES, AND 8 DEPLETED BURNABLE POISON RODS IN ONE OF THE REGION 8B ASSEMBLIES. THREE SECONDARY SOURCES WILL BE USED AS SHOWN IN FIGURE 1.

2.3 THERMAL AND HYDRAULIC DESIGN

NO SIGNIFICANT VARIATIONS IN THERMAL MARGINS WILL RESULT FROM THE CYCLE 7 RELOAD. THE PRESENT DNB CORE LIMITS (REFERENCE 7) HAVE BEEN FOUND TO BE CONSERVATIVE FOR CYCLE 7

3.0 POWER CAPABILITY AND ACCIDENT EVALUATION

3.1 POWER CAPABILITY

THE PLANT POWER CAPABILITY IS EVALUATED CONSIDERING THE CONSEQUENCES OF THOSE INCIDENTS EXAMINED IN THE FSAR (REF. 2) USING THE PREVIOUSLY ACCEPTED DESIGN BASIS. IT IS CONCLUDED THAT THE CORE RELOAD WILL NOT ADVERSELY AFFECT THE ABILITY TO SAFELY OPERATE AT 100 PERCENT OF RATED POWER DURING CYCLE 7. FOR THE EVALUATION PERFORMED TO ADDRESS OVERPOWER CONCERNS, THE FUEL CENTERLINE TEMPERATURE LIMIT OF 4700-F CAN BE ACCOMMODATED WITH MARGIN IN THE CYCLE 7 CORE USING THE METHODOLOGY DISCUSSED IN REFERENCE 1. THE TIME DEPENDENT DENSIFICATION MODEL (REF. 8) WAS USED FOR THESE FUEL TEMPERATURE EVALUATIONS. THE LOCA LIMIT AT RATED POWER CAN BE MET BY MAINTAINING FQ AT, OR BELOW, 2.18.

3.2 ACCIDENT EVALUATION

THE EFFECTS OF THE RELOAD ON THE DESIGN BASIS AND POSTULATED INCIDENTS ANALYZED IN THE FSAR (REF. 2) WERE EXAMINED. IN ALL CASES, IT WAS FOUND THAT THE EFFECTS WERE ACCOMMODATED WITHIN THE CONSERVATISM OF THE ASSUMPTIONS USED IN THE PREVIOUSLY APPLICABLE SAFETY ANALYSES.

A CORE RELOAD CAN TYPICALLY AFFECT ACCIDENT ANALYSIS INPUT PARAMETERS IN THE FOLLOWING AREAS: CORE KINETIC CHARACTERISTICS, CONTROL ROD WORTHS, AND CORE PEAKING FACTORS. CYCLE 7 PARAMETERS IN EACH OF THESE THREE AREAS WERE EXAMINED AS DISCUSSED BELOW TO ASCERTAIN WHETHER NEW ACCIDENT ANALYSES WERE REQUIRED.

3.2.1 KINETICS PARAMETERS

A SUMMARY OF THE EVALUATION OF CYCLE 7 CORE PHYSICS PARAMETERS WITH CURRENT LIMITS IS GIVEN IN TABLE 2. THE DELAYED NEUTRON FRACTIONS, MODERATOR TEMPERATURE COEFFICIENTS, AND PROMPT NEUTRON LIFETIME ARE WITHIN THE BOUNDS OF THE CURRENT LIMITS. THE MODERATOR TEMPERATURE COEFFICIENT WILL BE ZERO OR NEGATIVE DURING NORMAL OPERATION, ALTHOUGH OPERATION WITH A SLIGHTLY POSITIVE COEFFICIENT IS ALLOWED BELOW FULL POWER OPERATION. THE MOST NEGATIVE DOPPLER TEMPERATURE COEFFICIENT IS -2.3 PCM/-F COMPARED TO THE LIMIT OF -1.6 PCM/-F. THIS COEFFICIENT IS USED IN CONJUNCTION WITH THE DOPPLER POWER COEFFICIENT FOR FUEL TEMPERATURE CHANGES IN TRANSIENTS WHERE THE CORE WATER TEMPERATURE DROPS. FOR THE MOST SEVERE REACTIVITY ADDITION ACCIDENT (STARTUP OF AN INACTIVE LOOP), THIS AMOUNTS TO LESS THAN A 3% INCREASE IN TOTAL POSITIVE REACTIVITY INSERTION. THIS WOULD YIELD A NEGLIGIBLE INCREASE IN PEAK POWER WHICH CAN BE ACCOMMODATED IN ALL OF THE FSAR COOLDOWN EVENTS.

3.2.2 CONTROL ROD WORTHS

CHANGES IN CONTROL ROD WORTH MAY AFFECT DIFFERENTIAL ROD WORTHS, SHUTDOWN MARGIN, EJECTED ROD WORTHS, AND TRIP REACTIVITY. TABLE 2 SHOWS THAT THE MAXIMUM DIFFERENTIAL ROD WORTH OF TWO RCCA CONTROL BANKS MOVING TOGETHER IN THEIR HIGHEST WORTH REGION FOR CYCLE 7 MEETS THE CURRENT LIMIT. TABLE 3 SHOWS THAT THE CYCLE 7 SHUTDOWN MARGIN REQUIREMENTS ARE SATISFIED. EJECTED ROD WORTHS FOR CYCLE 7 ARE WITHIN THE BOUNDS OF THE CURRENT LIMITS.

AS A CONDITION FOR USING THE ROD SWAP TECHNIQUE FOR MEASURING ROD WORTHS, THE NRC HAS REQUIRED THAT A COMPARISON BE MADE BETWEEN WESTINGHOUSE AND VEPCO SHUTDOWN MARGIN CALCULATIONS (REFERENCE 9). THIS COMPARISON IS WILL BE PERFORMED AND REVIEWED BY THE STATION NUCLEAR SAFETY AND OPERATING COMMITTEE AND BY THE SAFETY EVALUATION AND CONTROL STAFF PRIOR TO UNIT 1 STARTUP.

3.2.3 CORE PEAKING FACTORS

THE PEAKING FACTORS FOR THE STEAMLINE BREAK HAVE BEEN EVALUATED AND ARE WITHIN THE BOUNDS OF THE PREVIOUS SAFETY ANALYSIS LIMITS. IN ADDITION, THE PEAKING FACTORS FOLLOWING CONTROL ROD EJECTION ARE WITHIN THE LIMITS OF PREVIOUS ANALYSIS VALUES FOR ALL CASES.

AS STATED IN SECTION 2.2, THE MAXIMUM ANALYTICALLY PREDICTED LOCAL PEAKING FACTOR FOR CYCLE 7 IS LESS THAN THE FQ LIMIT. THEREFORE, FREQUENT AXIAL POWER DISTRIBUTION MONITORING WILL NOT BE REQUIRED DURING CYCLE 7. IT IS ANTICIPATED THAT A CHANGE IN THE F-DELTA-H PART POWER MULTIPLIER FROM 0.2 TO 0.3 WILL BE REQUESTED IN ORDER TO RESTORE THE ROD INSERTION LIMITS TO THE CURRENTLY APPROVED VALUES UPON COMPLETION OF APPROXIMATELY 1000 MWD/MTU OF CYCLE LIFETIME.

4.0 TECHNICAL SPECIFICATIONS CHANGES

AS DISCUSSED IN SECTION 2.2, A CHANGE TO THE CONTROL ROD INSERTION LIMITS OF THE TECHNICAL SPECIFICATIONS IS BEING PROPOSED FOR CYCLE 7 OPERATION. THE REVISED LIMITS, PRESENTED IN FIGURE 2, REPRESENT A DECREASE (SHALLOWER INSERTION) OF 3% OF FULL ROD TRAVEL AT HOT FULL POWER AND 20% AT HOT ZERO POWER WITH RESPECT TO THE CURRENTLY APPROVED INSERTION LIMITS.

5.0 REFERENCES

1. F. M. BORDELON, ET AL.," WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," WCAP-9272, MARCH 1978.

2. SURRY POWER STATION UNIT 1 AND 2, FINAL SAFETY ANALYSIS REPORT, DECEMBER 1, 1969.

3. MILLER, J. V., (ED), "IMPROVED ANALYTICAL MODELS USED IN WESTINGHOUSE FUEL ROD DESIGN COMPUTATIONS", WCAP-8785, OCTOBER 1976.

4. GEORGE, R. A., ET AL.,"REVISED CLAD FLATTENING MODEL", WCAP-8377, JULY 1974.

5. RISHER, D. H., ET AL.,"SAFETY ANALYSIS FOR THE REVISED FUEL ROD INTERNAL PRESSURE DESIGN BASIS," WCAP-8964, JUNE 1977.

6. JONES, R. G., IORII, J. A., "OPERATIONAL EXPERIENCE WITH WESTINGHOUSE CORES", WCAP-8183, REVISION 11, MAY 1982.

7. SURRY POWER STATION UNITS 1 AND 2, TECHNICAL SPECIFICATIONS, DOCKET NOS. 50-280 AND 50-281, AS AMENDED.

8. HELLMAN, J. M.(ED.), "FUEL DENSIFICATION EXPERIMENTAL RESULTS AND MODEL FOR REACTOR OPERATION", WCAP-8219-A, MARCH 1975.

9. LETTER FROM R. L. TEDESCO (NRC) TO W. N. THOMAS (VEPCO), DATED NOVEMBER 7, 1980.

TABLE 1
FUEL ASSEMBLY DESIGN PARAMETERS
SURRY UNIT 1 CYCLE 7

BATCH	4C*	S2/6B	6C**	7A	7B	8A	8B	9***
ENRICHMENT (W/O U235)	3.33	3.20	2.90	2.90	3.39	3.22	3.40	3.59
DENSITY (% THEORETICAL)	94.40	94.48	94.30	94.50	94.70	94.61	94.58	94.60
NUMBER OF ASSEMBLIES	22	2	8	10	6	12	33	64
BURNUP AT BEGINNING OF CYCLE 7 (MWD/MTU)1	26200	27300	25100	24300	30000	20100	18000	0
BURNUP AT END OF CYCLE 7 (MWD/MTU)2	35900	32700	37700	32600	42600	31000	32600	14700
MTU PER REGION3	10.02	0.92	3.64	4.56	2.75	5.50	15.08	29.38

* FROM SURRY 1 CYCLE 4
** FROM SURRY 1 CYCLE 5
*** INCLUDES 4 FRESH ASSEMBLIES FROM SURRY 2 BATCH 9. WITH NOMINAL ENRICHMENT AND DENSITY OF 3.60% AND 94.5 W/O RESPECTIVELY.

1. ASSUME END-OF-CYCLE 6 BURNUP OF 16500 MWD/MTU
2. ASSUME END-OF-CYCLE 7 BURNUP OF 13000 MWD/MTU; ALL BATCH AVERAGE BURNUPS ARE <37000 MWD/MTU.
3. INITIAL MTU

TABLE 2
KINETICS CHARACTERISTICS
SURRY 1 CYCLE 7

PARAMETER	CURRENT LIMIT	CYCLE 7 VALUES
MODERATOR TEMPERATURE COEFFICIENT (PCM/-F)[1]	3.0 TO -35.0	WITHIN CURRENT LIMITS
MOST NEGATIVE DOPPLER-ONLY TEMPERATURE COEFFICIENT (PCM/-F)	-1.6	-2.3 [2]
LEAST NEGATIVE DOPPLER-ONLY POWER COEFFICIENT, HZP TO HFP (PCM/% POWER)	-11.4 TO -6.00	WITHIN CURRENT LIMITS
MINIMUM DELAYED NEUTRON FRACTION, BOL TO EOL (%)	0.55 TO 0.44	WITHIN CURRENT LIMITS
MAXIMUM PROMPT NEUTRON LIFETIME (MICRO SEC)	26	<26
MAXIMUM DIFFERENTIAL ROD WORTH OF TWO BANKS MOVING TOGETHER (PCM/SEC)	75	<75

1. 1 PCM = 10-5 DK/K
2. SEE SECTION 3.2.1

TABLE 3
SHUTDOWN REQUIREMENTS AND MARGINS
SURRY 1 CYCLE 7

CONTROL ROD WORTHS (% DK/K)	BOC	EOC
ALL RODS INSERTED LESS WORST STUCK ROD(1)	5.83	6.79
(1) LESS 10% (2)	5.25	6.11
CONTROL ROD REQUIREMENTS (% DK/K)		
REACTIVITY DEFECTS (COMBINED DOPPLER, TAVE, VOID, AND REDISTRIBUTION EFFECTS)	1.98	3.35
ROD INSERTION ALLOWANCE	1.00	0.50
TOTAL REQUIREMENTS (3)	2.98	3.85
SHUTDOWN MARGIN ((2)-(3))	2.27	2.26
REQUIRED SHUTDOWN MARGIN	1.77	1.77