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VICE PRESIDENT
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May 29, 1981

Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

ATTENTION: Mr. R. A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit No. 1 and Unit No. 2
Docket Nos. 50-318 and 50-319
Reanalysis of CEA Ejection Event Against New Criteria

REFERENCE (A): R. A. Clark to A. E. Lundvall letter dated 12/12/80

(B): R. A. Clark to A. E. Lundvall letter dated 2/10/81

Gentlemen:

References (A) and (B) issued amendments to the operating licenses. A condition of those amendments was the submittal of a reanalysis of the CEA Ejection Event. The purpose of the reanalysis was to calculate the number of fuel rods predicted to experience DNB during the event, show that the radiological consequences are within the guidelines of 10 CFR 100, and to demonstrate that the RCS upset pressure limit of 2750 psia is not exceeded. That reanalysis has been performed for Unit 1, Cycle 5 and Unit 2, Cycle 4. The attachment is a discussion of the reanalysis and its results. The results meet the new criteria.

Very truly yours,

BALTIMORE GAS AND ELECTRIC COMPANY

A. E. Lundvall, Jr.
Vice President - Supply

WJL/AEL/djw
Attachment

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ATTACHMENT

Introduction

The CEA Ejection event was reanalyzed to 1) calculate the number of fuel pins predicted to experience DNB during the event and show that, under the assumption that fuel which experiences DNB fails, the radiological consequences are within the guidelines of 10CFR Part 100, 2) demonstrate that the RCS upset pressure limit of 2750 psia is not exceeded. The following reanalysis is applicable to Calvert Cliffs Unit I Cycle 5 and Unit II Cycle 4.

An ejected CEA (control element assembly) is assumed to occur due to a complete circumferential break of either the control element drive mechanism (CEDM) housing or the CEDM nozzle of the reactor vessel. The ejection of the CEA inserts positive reactivity into the core. The addition of positive reactivity causes a rapid rise in power and an increasing heat flux. When the power excursion reaches the high power trip setpoint, a reactor trip is initiated. The Doppler fuel temperature coefficient mitigates the power rise while the heat flux continues to increase. Insertion of negative reactivity due to scram rod motion terminates the power increase and causes the heat flux to decrease. The decreasing heat flux terminates the departure from nucleate boiling (DNB).

Assumptions and Input Parameters

1. A BOC Doppler coefficient was used since it produces the least amount of negative reactivity feedback. A 15% allowance for uncertainties is included.

2. A BOC Moderator Temperature Coefficient (MTC) of $+.5 \times 10^{-4} \Delta\rho/°F$ was used since a positive MTC in conjunction with increasing coolant temperature results in positive feedback.

3. An EOC delayed neutron fraction was used to produce the highest power rise during the event.

4. The full and zero power cases were analyzed assuming an ejection time of .05 seconds which is consistent with the FSAR and previous reload submittals.

5. The high power trip setpoints for full and zero power ejections were set at 112% and 40% of 2754 MWt respectively.

6. Although C-E does not equate onset of departure from nucleate boiling (DNB) with clad failure, the analysis conservatively assumed that all fuel rods experiencing DNB fail.

7. The analysis conservatively assumed that 10% of the fuel pin activity is located in the fuel clad gap and that all the activity in the fuel clad gap is released to the coolant once the pin experiences DNB.

8. The entire primary coolant inventory is assumed to be dumped into containment.

9. The leak rate, L, from the containment is the maximum technical specification allowed value of 0.20% per day.

10. Activity released from the secondary system is based on the assumption that activity leaks from the primary to the secondary through the steam generators and escapes through the atmospheric dump valves. The leakage rate of the active primary coolant into the secondary assumed in this analysis is the maximum Technical Specification value of 1 GPM. The steam generator partition factor used is 0.10.

11. A value of $3.47X10^{-4}$ m^3/sec was assumed in the analysis for breathing rate. This is consistent with the FSAR value.

12. An atmospheric dispersion coefficient, χ/Q value of $1.8X10^{-4}$ sec/m^3 was assumed in the analysis consistent with the FSAR.

13. The peak pressure is conservatively calculated assuming the RCS pressure boundary is not breached.

The key input parameters used in the analysis are given in Table A-1.

Method of Analysis

The analytical method used in the reanalysis of this event is consistent with the NRC approved C-E CEA Ejection method described in Reference 1 except for the incorporation of DNB as the criteria for fuel failure.

The DNBR calculations were performed using the STRIKIN code (Reference 2) which models the fuel pin in detail to determine the transient heat flux and power. The static open channel thermal-hydraulic code, TORC (Reference 3) in combination with the CE-1 correlation was used to calculate the transient minimum DNBR. The procedure employed is to first calculate minimum DNBR parametric in pre- and post-ejected radial peaks. Radial peaking census information for the ejection event is then employed to establish the number of fuel pins which exceed the specified DNBR limit. A general outline of the methods used is given below.

1. With the maximum ejected worth and an assumed pre-ejected peak and post-ejected radial peak, simulate the CEA Ejection event with STRIKIN to obtain the transient heat flux values.

2. Input to TORC, the transient hot channel heat flux distributions, and the RCS pressure, temperature and mass flow rate to calculate minimum DNBR.

3. Repeat Steps 1 and 2 for different post-ejected radial peaks to obtain the post-ejected radial peak which results in a minimum CE-1 DNBR equal to 1.19 for the pre-ejected peak assumed in Step 1.

4. Repeat Steps 1 through 3 for a set of pre-ejected peaks to obtain a corresponding set of post ejected peaks which results in a DNBR of 1.19.

5. From the results of Steps 3 and 4, plot for each pre-ejected radial peak the corresponding post ejected radial peak which resulted in a DNBR of 1.19, as illustrated in Figure a.

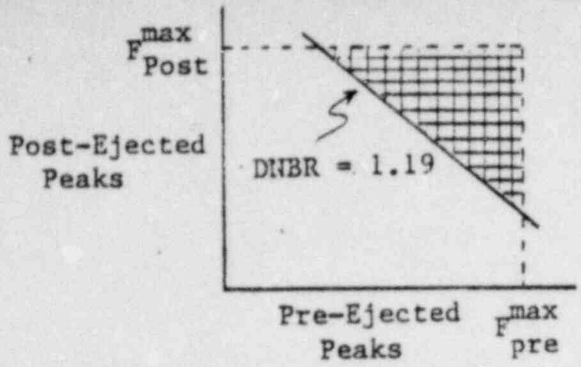


Figure a

The iso-DNBR curve of Figure a divides the "peak space" into a region corresponding to DNBR < 1.19 (crosshatched area) and a region corresponding to DNBR > 1.19. Any fuel rod whose "peak coordinates" are in the crosshatched area of Figure a will be assumed to experience DNB and thus clad failure.

6. Count the number of pins in the crosshatched area of Figure a using the pin census corresponding to the configuration being analyzed. The fuel pin census is obtained from two-dimensional fine mesh PDQ-7 calculations.

7. Determine the total iodine activity. The 0 - 2 hour I-131 site boundary dose is calculated from

$$DOSE\ (REM) = CF * B * \chi/Q\ [S_{Primary} * L + S_{Secondary} * PF]$$

where

$S_{Primary}$ = S_0 (initial activity in the primary coolant) + S_1 (activity released to containment by the failed fuel rods), in curies,

$S_{Secondary}$ = S_2 (activity leaked from primary to secondary in 2 hours), in curies,

B = breathing rate for the first 8 hours of a person offsite in m^3/sec,

χ/Q = atmospheric diffusion factor at the nearest exclusion zone boundary in sec/m^3,

L = leak rate from containment, in %/2 hrs

CF = $1.486X10^6$ rem/Ci, a conversion factor,

PF = steam generator partition factor.

8. Determine the total noble gas activity. The whole body dose is calculated using the following equation.

$$\text{Whole Body Dose (rem)} = [.25\ (\overline{E}_\gamma + \frac{.23}{.25}\overline{E}_\beta)] * \chi/Q * A_{total}$$

where

$\overline{E}_\beta$ = average energy release by beta decay, in Mev,

$\overline{E}_\gamma$ = average energy release by gamma decay, in Mev,

χ/Q = atmospheric diffusion factor at the nearest exclusion zone boundary, in sec/m^3,

A_{total} = A_0 (initial noble gas activity) + A_1 (activity released by the failed fuel rods) + A2 (activity leaked from primary to secondary), in curies.

Results and Conclusions

The results of the full and zero power CEA Ejection are presented in Table A-2. The dynamic response of the NSSS are presented in Figures A-1 and A-2.

The peak pressure of 2477 psia is below the pressure upset limit of 110% (2750 psia) of design pressure.

The number of fuel rods experiencing DNB is 10.7% for the ejection from full power and 6.6% for ejection from zero power. A maximum site boundary dose of 76.8 rem (DEQ I-131) and 0.25 rem (DEQ Xe-133) occurs for the full power ejection.

Since the site boundary 0 - 2 hr doses are within 10CFR Part 100 limits, and since the RCS pressure does not exceed 110% of design, it is concluded that the consequences of the CEA Ejection event are acceptable.

TABLE A-1

KEY PARAMETERS ASSUMED IN THE CEA EJECTION ANALYSES

Parameter	Units	Values
Full Power		
Core Power Level	MWt	2754
Core Average Linear Heat Generation Rate at 2754 MWt	KW/ft	6.52
Moderator Temperature Coefficient, MTC	$X10^{-4} \Delta\rho/°F$	± 0.50
Ejected CEA Worth	$\%\Delta\rho$	0.22
Delayed Neutron Fraction, β		.0044
Maximum Post Ejected Radial Power Peak		3.15
Axial Power Peak		1.35
CEA Bank Worth at Trip	$\%\Delta\rho$	-3.00
High Power Trip Delay Time	sec	0.90
Tilt Allowance		1.03
Doppler Multiplier		0.85
CEA Drop Time	sec	3.10
High Power Trip Setpoint	% of 2754 MWt	112
Zero Power		
Core Power Level	MWt	1.00
Ejected Worth	$\%\Delta\rho$	0.63
Maximum Post Ejected Radial Power Peak		9.40
Axial Power Peak		1.75
CEA Bank Worth at Trip	$\%\Delta\rho$	-1.50
Tilt Allowance		1.10
Doppler Multiplier		0.85
High Power Trip Setpoint	% of 2754 MWt	40

TABLE A-2

CEA EJECTION EVENT RESULTS

Parameter	Units	Value
Full Power		
Total Number of Fuel Rods with CE-1 DNBR $\leq$ 1.19	%	10.7
Peak Pressure	psia	2477
0 - 2 Hr Site Boundary Thyroid Dose	rem	76.8
0 - 2 Hr Site Boundary Whole Body Exposure	rem	0.25
Zero Power		
Total Number of Fuel Rods With CE-1 DNBR $\leq$ 1.19	%	6.6
Peak Pressure	psia	2455
0 - 2 Hr Site Boundary Thyroid Dose	rem	47.4
0 - 2 Hr Site Boundary Whole Body Exposure	rem	.16

References

1. CENPD-190A, CEA Ejection, C-E Method of Control Element Assembly Ejection," January, 1976.

2. CENPD-135, "STRIKIN II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," April, 1974, (Proprietary).

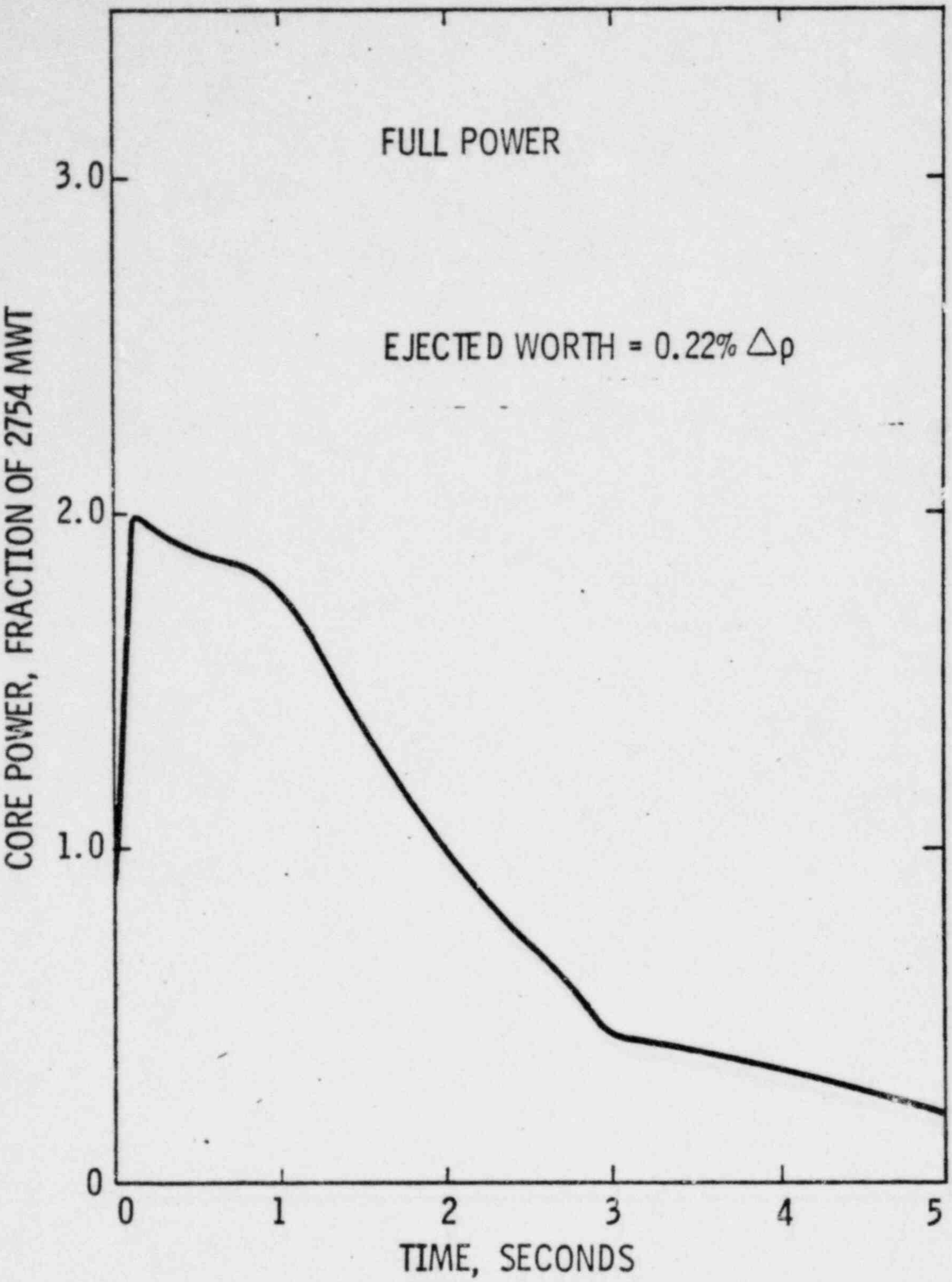
. CENPD-135, Supplement 2, "STRIKIN II, A Cylindrical Geometry Fuel Rod Heat Transfer Program (Modification)," February, 1975, (Proprietary).

CENPD-135, Supplement 4, "STRIKIN II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," August, 1976, (Proprietary).

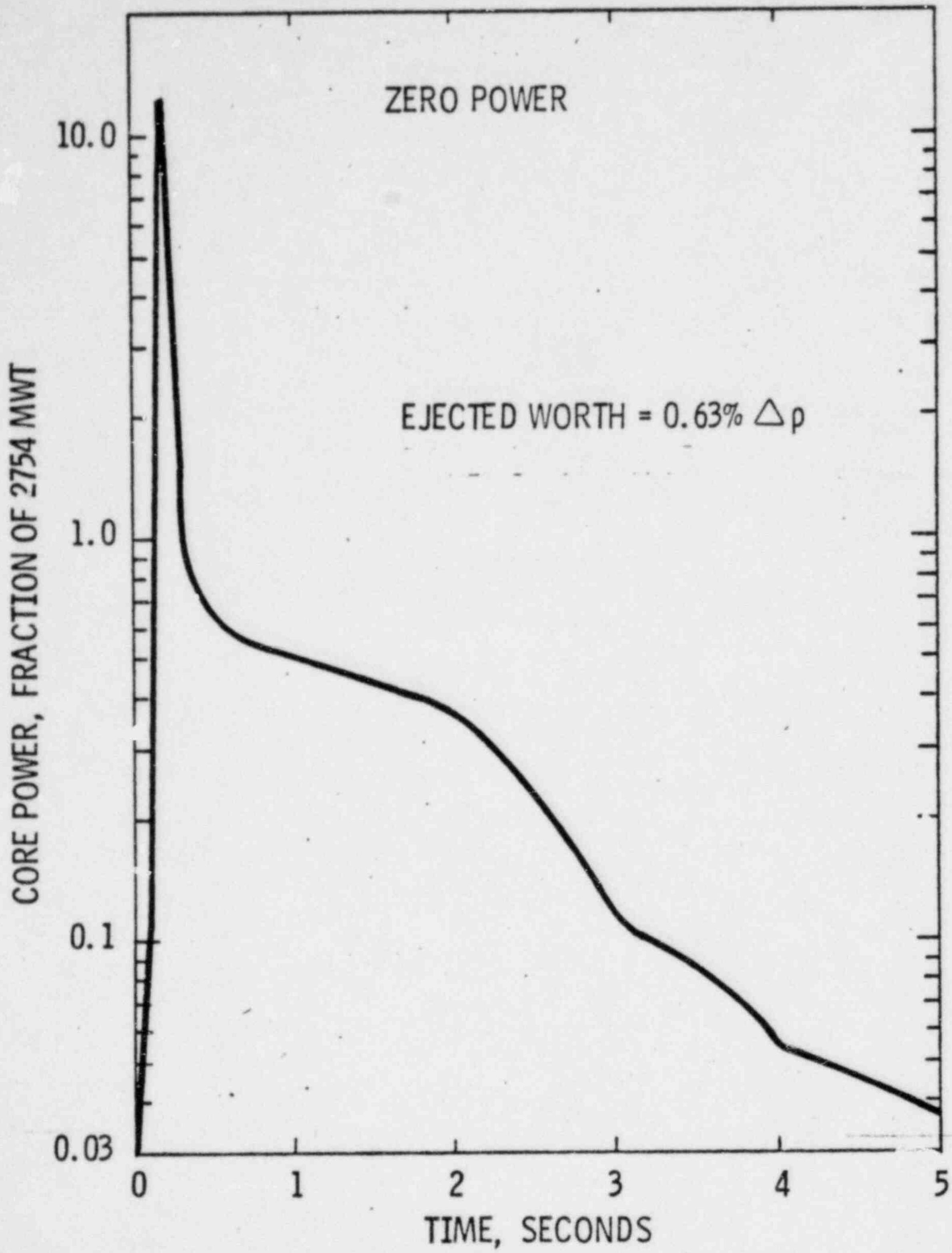
CENPD-135, Supplement 5, "STRIKIN II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," April, 1977, (Proprietary).

3. CENPD-161-P, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," July, 1975, (Proprietary).

CENPD-206-P, "TORC Code, Verification and Simplified Modeling Methods," January, 1977, (Proprietary).



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CEA EJECTION EVENT CORE POWER vs TIME	Figure A-1



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CEA EJECTION EVENT CORE POWER vs TIME	Figure A-2