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CHRIS H. POINDEXTER
VICE PRESIDENT
ENGINEERING AND CONSTRUCTION

August 30, 1985

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

ATTENTION: Mr. E. J. Butcher, Jr., Acting Chief
Operating Reactors Branch #3
Division of Licensing

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit No. 2; Docket No. 50-318
Request for Amendment to Operating License DPR-69
Seventh Cycle License Application

REFERENCE: A. E. Lundvall, Jr. to J. R. Miller letter dated February 22, 1985, Calvert Cliffs Nuclear Power Plant Unit 1; Docket No. 50-317, Amendment to Operating License DPR-53, Eighth Cycle License Application

ENCLOSURE: Unit 2 Cycle 7 Reload License Submittal

Gentlemen:

The Baltimore Gas and Electric Company hereby requests an Amendment to its Operating License No. DPR-69 for Calvert Cliffs Unit No. 2 to allow operation for a seventh cycle. The enclosed report presents a detailed description of the required Standard Technical Specifications (STS) with supporting safety analysis information to ensure conservative operations at a rated thermal power of 2700 MWth for Unit 2 Cycle 7.

Our present intention is to begin the Unit 2 refueling outage on October 18, 1985, and to complete the outage and begin the approach to criticality on December 9, 1985, with return to power operations immediately thereafter.

W/Check $150.00 #348084

Add: NRR/DSI/CPB
RGN2/DRSS/EPRPB
RM/DDAMI/MIB

A001
3/40

PROPOSED CHANGES (BG&E FCR 85-3003)

This reload is enveloped by the analysis previously submitted for the current operating Unit 1 Cycle 8 (Reference). The proposed changes to the Unit 2 Technical Specifications will make the following Technical Specifications identical to the Unit 1 Technical Specifications: STS 3.1.1.1, decreases the required shutdown margin; STS 3.1.1.4, changes the allowed Moderator Temperature Coefficient (MTC); STS 4.2.1.4, removes the flux peaking augmentation factors. These changes are all enveloped by the analysis performed for the referenced cycle and the reconfiguration of Control Element Assemblies (CEAs). One Technical Specification change that will be different from Unit 1 is Technical Specification B.2.1.1 which changes the minimum DNBR limit from 1.23 to 1.21. This is consistent with the Departure from Nucleate Boiling Ratio (DNBR) limit for the CE-1 correlation which the NRC recently approved.

DETERMINATION OF SIGNIFICANT HAZARD

We have determined, based on the analytical information supplied in the enclosure, that this amendment request does not involve a significant hazards consideration. The proposed changes are consistent with example iii of amendments considered not likely to involve significant hazards consideration, as shown in the Federal Register dated April 6, 1983, page 14870. No fuel assemblies to be loaded into the Unit 2 Cycle 7 core will be of new or different design than those used previously and found to be acceptable to the NRC. No proposed changes to the Technical Specifications for Unit 2 Cycle 7 involve acceptance criteria which are significantly different from those previously found acceptable to the NRC. The analytical methods used to determine conformance with the Technical Specifications and regulations are consistent with previous NRC approvals and involve no significant changes.

We conclude that the proposed reload license amendment does not involve a significant hazard consideration in that:

1. The probability or consequences of an accident evaluated is not significantly increased. The large break Loss of Coolant Accident was re-evaluated and its consequences were found to be slightly worse than previously reported, but still well below the license limit.

2. The reload application does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The license reload does not involve a significant reduction in the margin for safety. Small changes in shutdown margin, increase in moderator temperature coefficients, and decrease in DNBR ratio are compensated by the CEA reconfiguration and reference cycle analysis.

SAFETY COMMITTEE REVIEW

These proposed changes to the Technical Specifications and our determination of significant hazards have been reviewed by our Plant Operations and Off-Site Safety Review Committees, and they have concluded that implementation of these changes will not result in an undue risk to the health and safety of the public.

FEE DETERMINATION

Pursuant to 10 CFR 170.21, we are including BG&E Check Number A348084 in the amount of $150.00 to the NRC to cover the application fee for this request.

Very truly yours,

CHPoindexter

STATE OF MARYLAND:
: TO WIT
CITY OF BALTIMORE:

Chris H. Poindexter, being duly sworn states that he is Vice President of the Baltimore Gas and Electric Company, a corporation of the State of Maryland; that he provides the foregoing response for the purposes therein set forth; that the statements made are true and correct to the best of his knowledge, information, and belief; and that he was authorized to provide the response on behalf of said Corporation.

WITNESS my Hand and Notarial Seal: Marilyn R Cox
Notary Public

My Commission Expires: 7/1/86

CHP/JAM/lmt

cc: D. A. Brune, Esquire
G. F. Trowbridge, Esquire
D. H. Jaffe, NRC
T. Foley, NRC
T. Magette, DNR

ATTACHMENT TO

B-NE-

CALVERT CLIFFS UNIT 2 CYCLE 7

RELOAD LICENSE SUBMITTAL

Calvert Cliffs Unit 2 Cycle 7
Reload License Submittal

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Section

1.0 INTRODUCTION AND SUMMARY

This report provides an evaluation of design and performance for the operation of Calvert Cliffs Unit 2 during its seventh fuel cycle, at full rated power of 2700 MWt. All planned operating conditions remain the same as those for Cycle 6. The core will consist of 148 presently operating Batch G and H assemblies, 60 fresh Batch J assemblies, 8 Batch E assemblies previously discharged from Cycle 4 of Calvert Cliffs Unit 2, and 1 Batch D assembly previously discharged from Cycle 3 of Calvert Cliffs Unit 2.

In addition to the plant operating requirements which ordinarily create a need for flexibility in the termination point of the previous cycle, the malfunction of the Unit 1 electric generator resulted in the need for additional flexibility in the form of an extended Cycle 6 length. Consequently, in place of the standard 1000 MWD/T end-of-cycle burnup range for the previous cycle, the Unit 2 Cycle 7 analyses have been performed for a Cycle 6 termination point between 12,200 MWD/T and 13,800 MWD/T.

In performing analyses of design basis events, determining limiting safety settings and establishing limiting conditions for operation, limiting values of key parameters were chosen to assure that expected Cycle 7 conditions would be enveloped, provided the Cycle 6 termination point falls within the expanded cycle burnup range discussed above. The analysis presented herein will accommodate a Cycle 7 length of up to 13,900 MWD/T.

The evaluations of the reload core characteristics have been conducted with respect to the Calvert Cliffs Unit 1 Cycle 8 safety analysis described in Reference 1, hereafter referred to as the "reference cycle" in this report unless otherwise noted. This is an appropriate reference cycle because of the similarity in the basic system characteristics of the two reload cores.

Specific core differences have been accounted for in the present analysis. In all cases, it has been concluded that either the reference cycle analyses envelope the new conditions or the revised analyses presented herein continue to show acceptable results. Where dictated by variations from the previous cycle (Unit 2 Cycle 6, Reference 2), proposed modifications to the plant Technical Specifications are provided and are justified by the analyses reported herein. These proposed modifications are very similar to those approved (Reference 3) for the reference cycle.

All Cycle 7 analyses address fuel exposure explicitly. The performance of Combustion Engineering 14x14 fuel at extended burnup is discussed in Reference 4.

2.0 OPERATING HISTORY OF THE PREVIOUS CYCLE

Calvert Cliffs Unit 2 is presently operating in its sixth fuel cycle utilizing Batch H, G, F, E and D fuel assemblies (including eight Batch E assemblies from Unit 1). Calvert Cliffs Unit 2 Cycle 5 began operation on June 29, 1984 and reached full power on July 24, 1984. The Cycle 6 startup testing was reported to the NRC in Reference 5. The reactor has operated up to the present time with the core reactivity, power distributions and peaking factors closely following the calculated predictions.

It is presently estimated that Cycle 6 will terminate on or about October 18, 1985. The Cycle 6 termination point can vary between 12,200 MWD/T and 13,800 MWD/T to accommodate the plant schedule and still be within the assumptions of the Cycle 7 analyses. As of August 19, 1985, the Cycle 6 burnup had reached 11,329 MWD/T.

3.0 GENERAL DESCRIPTION

The Cycle 7 core will consist of the number and types of assemblies and fuel batches as described in Table 3-1. The primary change to the core in Cycle 7 is the removal of 69 assemblies (61 Unit 2 assemblies: 40 Batch F, 1 Batch D, 20 Batch D/; 8 Unit 1 Batch E/ assemblies). These assemblies will be replaced by 40 fresh unshimmed Batch J assemblies (4.05 wt% U-235 enrichment), 20 fresh unshimmed Batch J* assemblies (3.40 wt% U-235 enrichment), 8 Batch E assemblies (3.03 wt% U-235) discharged from Unit 2 Cycle 4, and 1 Batch D assembly (3.03 wt% U-235) discharged from Unit 2 Cycle 3.

Figure 3-1 shows the fuel management pattern to be employed in Cycle 7. Figure 3-2 shows the locations of the poison pins within the lattice of twice-burned Batch G/ assemblies and the fuel rod locations in unshimmed assemblies. This fuel management pattern will accommodate Cycle 6 termination burnups from 12,200 MWD/T to 13,800 MWD/T.

The Cycle 7 core loading pattern is 90^{0} rotationally symmetric. That is, if one quadrant of the core were rotated 90^{0} into its neighboring quadrant, each assembly would be aligned with a similar assembly. This similarity includes batch type, number of fuel rods, initial enrichment and burnup.

Figure 3-3 shows the beginning of Cycle 7 assembly burnup distribution for a Cycle 6 termination burnup of 13,800 MWD/T. The initial enrichment of the fuel assemblies is also shown in Figure 3-3. Figure 3-4 shows the end of Cycle 7 assembly burnup distribution. The end of Cycle 7 core average exposure is approximately 29,700 MWD/T and the average discharge exposure is approximately 41,400 MWD/T. The end of cycle burnups are based on Cycle 6 and Cycle 7 lengths of 13,800 MWD/T and 13,900 MWD/T, respectively.

3.1 Prototype CEA

The prototype CEA is described in Reference 6. Cycle 3 was the first cycle of irradiation for this CEA. During the EOC-4 and EOC-5 outages this CEA was examined, as described in References 7 and 8, respectively. This prototype CEA was utilized in the center core position during Cycles 3 through 6. Due to the changes in CEA pattern which specify the use of a very weak CEA in the center location (see Subsection 3.2 below and the reference cycle submittal (Reference 1)), this CEA will be shifted to a new location.

3.2 CEA Patterns

The composition of nine CEAs, the configurations of two CEA banks and, consequently, the overall CEA bank pattern are being changed for Cycle 7. Both these changes and the reasons for them are identical to those presented in support of the reference cycle (Reference 1).

TABLE 3-1
CALVERT CLIFFS UNIT 2 CYCLE 7
CORE LOADING

Assembly Designation	Number of Assemblies	Initial Enrichment (wt% U-235)	Batch Burnup (MWD/T) BOC7(4)	Batch Burnup (MWD/T) EOC7(4,5)	Poison Rods Per Assembly	Initial Poison Loading (wt% B_4C)	Total Number of Poison and Non-Fuel Rods	Total Number of Fuel Rods
J	40	4.05	0	12,200	0	0	0	7040
J*	20	3.40	0	17,400	0	0	0	3520
H(1)	48	4.05	12,600	28,100	0	0	0	8448
H*(1)	24	3.40	16,900	31,100	0	0	0	4224
G(1)	48	4.00	27,400	40,900	0	0	0	8448
G*(1)	28	3.55	31,400	43,400	8	3.03	224	4704
E(2)	8	3.03	25,600	37,900	0	0	0	1408
D(3)	1	3.03	22,300	34,500	0	0	0	176
TOTAL	217		15,800	29,700			224	37,968

) Carried over from Cycle 6 to Cycle 7 of Unit 2

) Twice burned Batch E fuel discharged from Unit 2 Cycle 4

) Twice burned Batch D fuel discharged from Unit 2 Cycle 3

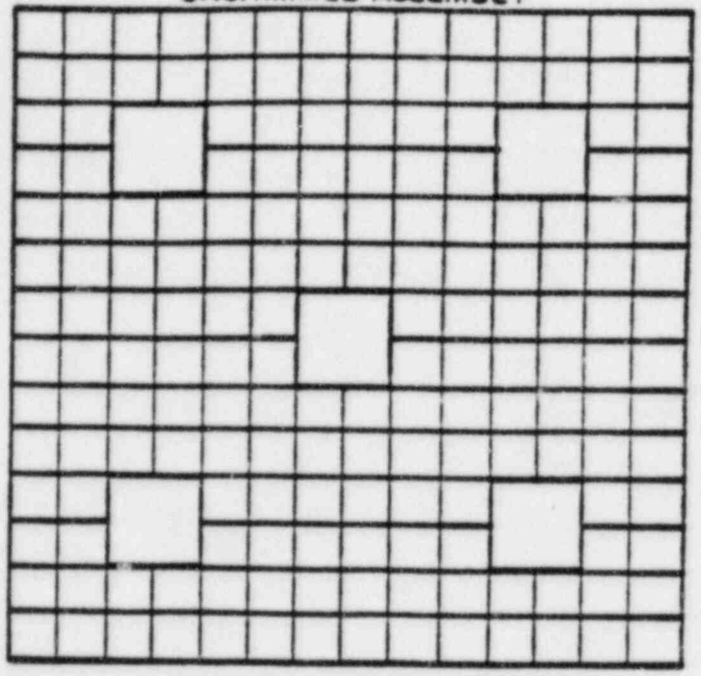
) Cycle 6 burnup of 13,800 MWD/T

) Cycle 7 burnup of 13,900 MWD/T

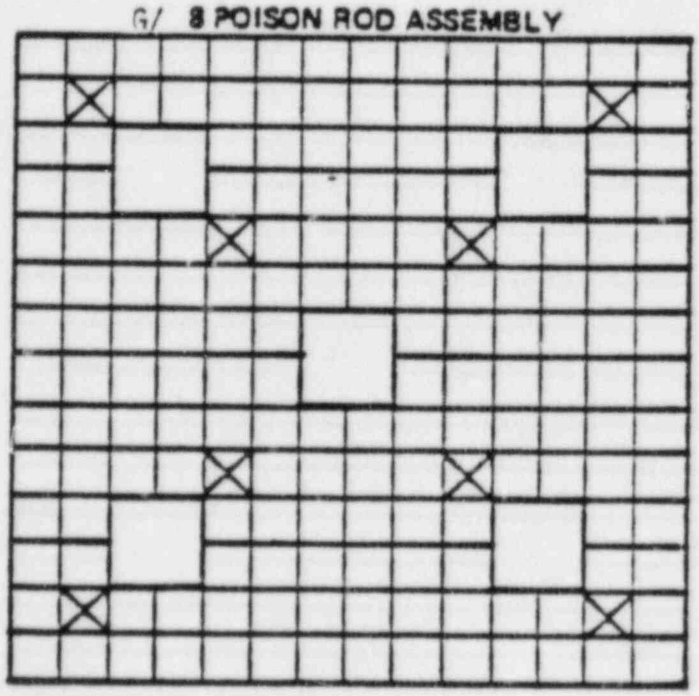
						1 J	2 J	
				3 J	4 J	5 H	6 G/	7 H
			8 J	9 H	10 G	11 H	12 G	13 G
		14 J	15 H*	16 H*	17 H	18 E	19 J*	20 H*
	21 J	22 H	23 H*	24 G	25 G	26 H	27 G/	28 H*
	29 J	30 G	31 H	32 G	33 J*	34 G/	35 J*	36 G/
45 J	37 H	38 H	39 E	40 H	41 G/	42 G	43 G	44 H*
54 J	46 G/	47 G	48 J*	49 G/	50 J*	51 G	52 H	53 G
	55 H	56 G	57 H*	58 H*	59 G/	60 H*	61 G	62 D

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 2 CYCLE 7 CORE MAP	Figure 3-1

UNSHIMMED ASSEMBLY



G/ 8 POISON ROD ASSEMBLY



FUEL ROD LOCATION

POISON ROD LOCATION

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 2 CYCLE 7 ASSEMBLY FUEL AND OTHER ROD LOCATIONS	Figure 3-2

INITIAL ENRICHMENT
BOC 7 BURNUP (MWD/T), EOC 6 = 13,800 MWD/T

						1 J 4.05 0	2 J 4.05 0	
				3 J 4.05 0	4 J 4.05 0	5 H 4.05 11,800	6 G/ 3.55 31,100	7 H 4.05 10,300
			8 J 4.05 0	9 H 4.05 10,700	10 G 4.00 26,200	11 H 4.05 14,200	12 G 4.00 30,100	13 G 4.00 29,200
		14 J 4.05 0	15 H* 3.40 16,300	16 H* 3.40 16,600	17 H 4.05 13,100	18 E 3.03 25,500	19 J* 3.40 0	20 H* 3.40 17,500
	21 J 4.05 0	22 H 4.05 10,700	23 H* 3.40 16,600	24 G 4.00 26,300	25 G 4.00 27,400	26 H 4.05 15,500	27 G/ 3.55 32,100	28 H* 3.40 16,950
	29 J 4.05 0	30 G 4.00 26,100	31 H 4.05 13,400	32 G 4.00 28,300	33 J* 3.40 0	34 G/ 3.55 32,400	35 J* 3.40 0	36 G/ 3.55 31,100
45 J 4.05 0	37 H 4.05 11,300	38 H 4.05 14,200	39 E 3.03 25,600	40 H 4.05 15,600	41 G/ 3.55 32,400	42 G 4.00 26,800	43 G 4.00 26,800	44 H* 3.40 17,450
54 J 4.05 0	46 G/ 3.55 28,900	47 G 4.00 30,100	48 J* 3.40 0	49 G/ 3.55 32,100	50 J* 3.40 0	51 G 4.00 26,700	52 H 4.05 10,200	53 G 4.00 25,150
	55 H 4.05 10,300	56 G 4.00 29,200	57 H* 3.40 17,500	58 H* 3.40 16,950	59 G/ 3.55 31,100	60 H* 3.40 17,450	61 G 4.00 25,150	62 D 3.03 22,300

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 2 CYCLE 7 ASSEMBLY AVERAGE BURNUP AT BOC AND INITIAL ENRICHMENT DISTRIBUTION	Figure 3-3

						1 J 10,100	2 J 13,500	
				3 J 11,200	4 J 14,300	5 H 25,300	6 G/ 42,200	7 H 25,800
			8 J 11,900	9 H 26,400	10 G 40,300	11 H 29,900	12 G 42,700	13 G 41,700
		14 J 11,900	15 H* 29,600	16 H* 31,300	17 H 29,700	18 E 37,900	19 J* 17,600	20 H* 31,950
	21 J 11,200	22 H 26,400	23 H* 31,200	24 G 40,100	25 G 41,400	26 H 31,000	27 G/ 44,400	28 H* 30,750
	29 J 14,300	30 G 40,300	31 H 29,900	32 G 42,000	33 J* 17,500	34 G/ 44,500	35 J* 17,200	36 G/ 43,400
	37 H 25,400	38 H 29,900	39 E 37,900	40 H 31,000	41 G/ 44,500	42 G 39,800	43 G 40,500	44 H* 31,600
45 J 10,200	46 G/ 40,300	47 G 42,700	48 J* 17,600	49 G/ 44,400	50 J* 17,300	51 G 40,600	52 H 26,900	53 G 39,250
54 J 13,500	55 H 25,800	56 G 41,700	57 H* 31,950	58 H* 30,750	59 G/ 43,400	60 H* 31,600	61 G 39,250	62 D 34,500

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 2 CYCLE 7 ASSEMBLY AVERAGE BURNUP AT EOC (MWD/T)	Figure 3-4

4.0 FUEL SYSTEM DESIGN

4.1 Mechanical Design

4.1.1 Fuel Design

The mechanical design for the Batch J reload fuel is identical to that of the Batch K fuel described in the reference cycle submittal (Calvert Cliffs Unit 1 Cycle 8, Reference 1). The mechanical designs of the Calvert Cliffs Unit 2 Batch H, G, E and D fuel assemblies were described in References 2, 3, 4 and 5, respectively.

4.1.2 Clad Collapse

A draft copy of an EPRI-sponsored report dealing with the phenomena of interpellet gap formation and clad collapse in modern PWR fuel rods was submitted to the NRC as Attachment 5 of Reference 6. The final version of this report was subsequently issued as Reference 7. A synopsis of the report focusing on C-E manufactured modern fuel was also submitted to the NRC, at the same time as the draft report, as Attachment 4 of Reference 6. Based upon the conclusion and recommendation of this synopsis that cycle specific clad collapse analyses are not necessary for modern C-E manufactured fuel, a cycle specific calculation was not prepared for the reference cycle. NRC concurrence with this approach was presented in Reference 8.

Consistent with the above discussion, a cycle specific clad collapse analysis was not prepared for Unit 2 Cycle 7. Since clad collapse has been effectively removed as an issue for modern C-E fuel provided the fuel design remains basically unchanged, discussion of clad collapse will be absent in subsequent license submittals where the fuel design is virtually the same as that of the reference cycle.

4.1.3 Dimensional Changes

All fuel assemblies in Cycle 7 with the exception of the eight Batch E fuel assemblies were reviewed for shoulder gap clearance using the SIGREEP model described in Reference 9 (approved in Reference 10). The shoulder gap clearance evaluation for the eight Batch E fuel assemblies was based upon their end of Cycle 7 fluences and the shoulder gap measurements taken on other fuel assemblies from the same batch of fuel. All fuel assemblies were reviewed for fuel assembly length clearance using the refined correlation discussed in References 11 and 12. All shoulder gap and fuel assembly length clearances were found to be adequate for Cycle 7.

4.1.4 CEA Design

The replacement CEAs to be utilized for the changes discussed in Section 3.3 have exactly the same design as those replacement CEAs installed in the reference cycle.

4.1.5 Removal of CEA Plugs

Unit 2 is presently operating with CEA plugs installed in the locations originally occupied by Part Length Rods (PLRs). These CEA plugs will be removed for Cycle 7 for exactly the same reasons and with exactly the same justification as presented in support of the reference cycle.

4.1.6 Metallurgical Requirements

The metallurgical requirements of the fuel cladding and the fuel assembly structural members for the Batch J fuel are identical to those of the reference cycle. Thus, the chemical or metallurgical performance of the Batch J fuel will be the same as that of the Unit 1 Cycle 8 fuel.

4.2 Hardware Modifications to Mitigate Guide Tube Wear

All standard fuel assemblies which will be placed in CEA locations in Cycle 7 will have stainless steel sleeves installed in the guide tubes to prevent guide tube wear. A detailed discussion of the design of the sleeves in irradiated fuel assemblies and their effect on reactor operation is contained in Reference 13. A modified short sleeve design (Reference 14) which is identical to that used in the reference cycle will be used in Batch J fuel assemblies.

4.3 Thermal Design

The thermal performance of a composite fuel pin which envelopes the various fuel assemblies present in Cycle 7 (fuel Batches J, H, G, E and D) has been evaluated using the FATES3 version of the fuel evaluation model (References 15 and 16), as approved by the NRC (Reference 17). The analysis was performed with a history that modeled the power and burnup levels representative of the peak pin at each burnup interval, from beginning of cycle to end of cycle burnups. The burnup range analyzed is in excess of that expected at end of Cycle 7.

The augmentation factor is being removed from the Tech. Specs. and the values of several items used in the Incore Monitoring System, i.e., measurement-calculational uncertainty, and axial fuel densification and thermal expansion factor, are being lowered (See Section 9.0). These changes and their effects are identical to those considered in the reference cycle.

5.0 NUCLEAR DESIGN

5.1 Physics Characteristics

5.1.1 Fuel Management

The Cycle 7 fuel management employs a mixed central region as described in Section 3, Figure 3-1. The fresh Batch J fuel is comprised of two sets of assemblies, each having a unique enrichment in order to minimize radial power peaking. There are 40 assemblies with an enrichment of 4.05 wt% U-235 and 20 assemblies with an enrichment of 3.40 wt% U-235. With this loading, the Cycle 7 burnup capacity for full power operation is expected to be between 12,600 MWD/T and 13,700 MWD/T, depending on the final Cycle 6 termination point. The Cycle 7 core characteristics have been examined for Cycle 6 terminations between 12,200 and 13,800 MWD/T and limiting values established for the safety analyses. The loading pattern (see Section 3) is applicable to any Cycle 6 termination point between the stated extremes.

Physics characteristics including reactivity coefficients for Cycle 7 are listed in Table 5-1 along with the corresponding values from the reference cycle (Reference 1). Please note that the values of parameters actually employed in safety analyses are different from those displayed in Table 5-1 and are typically chosen to conservatively bound predicted values with accommodation for appropriate uncertainties and allowances.

Table 5-2 presents a summary of CEA shutdown worths and reactivity allowances for the end of Cycle 7 zero power steam line break accident and a comparison to reference cycle data. The EOC zero power steam line accident was selected since it is the most limiting zero power transient with respect to reactivity requirements and, thus, provides the basis for verifying the Technical Specification required shutdown margin.

Table 5-3 shows the reactivity worths of the three CEA groups which are allowed in the core during critical/power conditions. These reactivity worths were calculated at full power conditions for Cycle 7 and the reference cycle. The configurations of CEA Groups 5 and 4 are being changed as discussed in Section 3; the new configurations will be identical to those of the reference cycle. The configuration of Group 3 which remains unchanged is also identical to that of the reference cycle. The power dependent insertion limit (PDIL) curve is the same as that of the reference cycle.

5.1.2 Power Distributions

Figures 5-1 through 5-3 illustrate the all rods out (ARO) planar radial power distributions at BOC7, MOC7 and EOC7, respectively, that are characteristic of the high burnup end of the Cycle 6 shutdown window. These planar radial power peaks are characteristic of the major portion of the active core length between about 20 and 80 percent of the fuel height. The high burnup end of the Cycle 6 shutdown window tends to increase the radial power peaking in this central axial region of the core for Cycle 7. The planar radial power distributions for the above region with CEA Group 5 fully inserted at beginning and end of Cycle 7 are shown in Figures 5-4 and 5-5, respectively, for the high burnup end of the Cycle 6 shutdown window.

The radial power distributions described in this section are calculated data without uncertainties or other allowances. However, the single rod power peaking values do include the increased peaking that is characteristic of fuel rods adjoining the water holes in the fuel assembly lattice. For both DNB and kw/ft safety and setpoint analyses in either rodded or unrodded configurations, the power peaking values actually used are higher than those expected to occur at any time during Cycle 7. These conservative values, which are used in Section 7 of this document, establish the allowable limits for power peaking to be observed during operation.

The range of allowable axial peaking is defined by the Limiting Conditions for Operation (LCOs) covering Axial Shape Index (ASI). Within these ASI limits, the necessary DNBR and kw/ft margins are maintained for a wide range of possible axial shapes. The maximum three-dimensional or total peaking factor anticipated in Cycle 7 during normal base load, all rods out operation at full power is 1.91, not including uncertainty allowances.

5.1.3 Safety Related Data

5.1.3.1 Ejected CEA Data

The Cycle 7 safety related data for this section are identical to the safety related data used in the reference cycle.

Subsequent to the submission of the reference cycle application (Reference 1), the maximum reactivity worth and planar power peak used in the analysis of the Ejected CEA Event at Hot Zero Power were revised. Table 5-4 which contains these data is included herein solely to update these values to those used in the revised analysis. Revision of these data did not affect the conclusions contained in Section 7.3.1 of Reference 1.

5.1.3.2 Dropped CEA Data

The Cycle 7 safety related data for this section are identical to the safety related data used in the reference cycle.

5.1.3.3 Augmentation Factors

A draft copy of an EPRI-sponsored report dealing with the phenomenon of interpellet gap formation in modern PWR fuel and a synopsis of this report focusing on only C-E manufactured modern fuel were submitted to the NRC as Attachments 5 and 4, respectively, of Reference 2. These documents presented the analyses which demonstrate that the increased power peaking associated with the small interpellet gaps found in modern, i.e, pre-pressurized and non-densifying, fuel is insignificant compared to the uncertainties in the safety analyses and Tech. Specs. Consequently, augmentation factors were eliminated from the reference cycle analysis and the Unit 1 Tech. Specs. NRC concurrence with this approach was presented in Reference 3.

Consistent with the above discussion, augmentation factors were eliminated from the Unit 2 Cycle 7 analysis and are being proposed for elimination in the Unit 2 Tech. Specs. (see Section 9). Since gap formation and the resulting need for augmentation factors has been effectively removed as an issue for modern C-E fuel provided the fuel design remains basically unchanged, discussion of augmentation factors will be absent in subsequent license submittals where the fuel design is virtually the same as that of the reference cycle.

5.1.3.4 Fuel Temperature Coefficient Bias

The Cycle 7 safety related data for this section are identical to the safety related data used in the reference cycle.

5.2 Analytical Input to In-Core Measurements

In-core detector measurement constants to be used in evaluating the reload cycle power distributions will be calculated in the same manner as those for the reference cycle.

5.3 Nuclear Design Methodology

Analyses have been performed in the same manner and with the same methodologies used for the reference cycle analyses.

5.4 Uncertainties in Measured Power Distributions

The power distribution measurement uncertainties to be applied to Cycle 7 are the same as those applied to the reference cycle.

TABLE 5-1

CALVERT CLIFFS UNIT 2 CYCLE 7
NOMINAL PHYSICS CHARACTERISTICS

	Units	Reference Cycle (Unit 1 Cycle 8)	Unit 2 Cycle 7
Dissolved Boron			
Hot Full Power, All Rods Out Equilibrium Xenon Boron Content for Criticality at BOC	PPM	1200	1090
Boron Worth			
Hot Full Power BOC	PPM/%$\Delta\rho$	107	107
Hot Full Power EOC	PPM/%$\Delta\rho$	88	87
Reactivity Coefficients (CEAs Withdrawn)			
Moderator Temperature Coefficients, Hot Full power, Equilibrium Xenon			
Beginning of Cycle	$10^{-4}\Delta\rho/^{o}F$	-0.1	-0.1
End of Cycle	$10^{-4}\Delta\rho/^{o}F$	-2.3	-2.3
Doppler Coefficient			
Hot Zero Power BOC	$10^{-5}\Delta\rho/^{o}F$	-1.56	-1.57
Hot Full Power BOC	$10^{-5}\Delta\rho/^{o}F$	-1.26	-1.25
Hot Full Power EOC	$10^{-5}\Delta\rho/^{o}F$	-1.44	-1.44
Total Effective Delayed Neutron Fraction, β_{eff}			
BOC		0.00604	0.00593
EOC		0.00516	0.00514
Neutron Generation Time, ℓ^*			
BOC	10^{-6} sec	23.1	23.1
EOC	10^{-6} sec	28.2	28.4

TABLE 5-2

CALVERT CLIFFS UNIT 2 CYCLE 7
LIMITING VALUES OF REACTIVITY WORTHS AND ALLOWANCES
FOR THE END-OF-CYCLE (EOC) HOT ZERO POWER (HZP)
STEAM LINE RUPTURE ACCIDENT, $\%\Delta\rho$

	Reference Cycle*	Unit 2 Cycle 7
1. Worth of all CEA's Inserted	9.3	8.7
2. Stuck CEA Allowance	2.5+	2.1
3. Worth of all CEA's less Worth of CEA Stuck Out**	6.8+	6.6
4. Power Dependent Insertion Limit CEA Bite at Zero Power	1.8	1.8
5. Calculated Scram Worth	5.0+	4.8
6. Physics Uncertainty plus Bias	0.7+	0.6
7. Net Available Scram Worth	4.3	4.2
8. Technical Specification Shutdown Margin	3.5	3.5
9. Margin in Excess of Technical Specification Shutdown Margin	0.8	0.7

*Unit 1 Cycle 8.

**Stuck CEA is one which yields worst results for EOC HZP SLB, i.e., worst combination of scram worth and reactivity insertion with cooldown.

+These are revised values; revision of these values was made soon after the submission of Reference 1 and did not affect the conclusion demonstrated in Table 5-2 of Reference 1.

TABLE 5-3

CALVERT CLIFFS UNIT 2 CYCLE 7
REACTIVITY WORTH OF CEA REGULATING
GROUPS AT HOT FULL POWER, $\%\Delta\rho$

	Beginning of Cycle		End of Cycle	
Regulating CEA's	Reference* Cycle	Unit 2 Cycle 7	Reference* Cycle	Unit 2 Cycle 7
Group 5	0.28	0.28	0.36	0.39
Group 4	0.82	0.83	0.91	0.88
Group 3	0.94	0.91	1.02	1.02

Note

Values shown assume sequential group insertion.

*Unit 1 Cycle 8

TABLE 5-4

CALVERT CLIFFS UNIT 2 CYCLE 7
CEA EJECTION DATA

	Limiting Values	
	Reference Cycle Safety Analysis Value*	Unit 2 Cycle 7 Safety Analysis Value
Maximum Radial Power Peak		
Full power with Bank 5 inserted; worst CEA ejected	3.6	3.6
Zero power with Banks 5+4+3 inserted; worst CEA ejected	8.9+	8.9
Maximum Ejected CEA Worth (% Δρ)		
Full power with Bank 5 inserted; worst CEA ejected	0.28	0.28
Zero power with Banks 5+4+3 inserted; worst CEA ejected	0.77+	0.77

* Unit 1 Cycle 8 (Reference 1)

+These are revised values; revision of these values was made soon after the submission of Reference 1 and did not affect the conclusions contained in Section 7.3.1 of Reference 1.

Notes

1. Uncertainties and allowances are included in the above data.
2. The safety analysis values are conservative with respect to the actual calculated values.

						1 0.72	2 1.00	
				3 0.84	4 1.08 X	5 0.99	6 0.78	7 1.14
			8 0.89	9 1.22	10 1.03	11 1.16	12 0.87	13 0.85
		14 0.89	15 0.98	16 1.07	17 1.24	18 0.85	19 1.30	20 1.03
	21 0.83	22 1.22	23 1.07	24 0.97	25 0.98	26 1.12	27 0.87	28 0.97
	29 1.07	30 1.03	31 1.22	32 0.96	33 1.28	34 0.84	35 1.24	36 0.86
45 0.71	37 0.99	38 1.15	39 0.84	40 1.11	41 0.84	42 0.88	43 0.94	44 0.98
54 0.98	46 0.79	47 0.86	48 1.29	49 0.86	50 1.26	51 0.96	52 1.18	53 0.94
	55 1.14	56 0.85	57 1.03	58 0.97	59 0.86	60 0.98	61 0.94	62 0.78

NOTE: X = MAXIMUM 1-P PEAK = 1.52

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 2 CYCLE 7 ASSEMBLY RELATIVE POWER DENSITY AT BOC EQUILIBRIUM XENON	Figure 5-1

						1: 0.74	2: 1.00	
				3: 0.81	4: 1.04	5: 0.99	6: 0.81	7: 1.15
			8: 0.87	9: 1.15	10: 1.00	11: 1.15	12: 0.89	13: 0.89
		14: 0.87	15: 0.95	16: 1.04	17: 1.20	18: 0.86	19: 1.28	20: 1.04
	21: 0.81	22: 1.15	23: 1.04	24: 0.96	25: 0.97	26: 1.13	27: 0.88	28: 0.99
	29: 1.04	30: 1.00	31: 1.19	32: 0.95	33: 1.25	34: 0.86	35: 1.25	36: 0.88
45: 0.75	37: 0.99	38: 1.15	39: 0.86	40: 1.12	41: 0.86	42: 0.92	43: 0.96	44: 1.01
	46: 0.83	47: 0.89	48: 1.29 X	49: 0.89	50: 1.26	51: 0.97	52: 1.20	53: 0.98
54: 1.00	55: 1.15	56: 0.89	57: 1.04	58: 0.99	59: 0.88	60: 1.01	61: 0.98	62: 0.84

NOTE: X = MAXIMUM 1-PIN PEAK = 1.45

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 2 CYCLE 7 ASSEMBLY RELATIVE POWER DENSITY AT 7 GWD/T EQUILIBRIUM XENON	Figure 5-2

							1 0.80	2 1.03 X
				3 0.82	4 1.04	5 1.01	6 0.86	7 1.17
			8 0.88	9 1.11	10 0.99	11 1.13	12 0.92	13 0.91
		14 0.88	15 0.94	16 1.01	17 1.16	18 0.88	19 1.25	20 1.02
	21 0.82	22 1.11	23 1.01	24 0.95	25 0.96	26 1.11	27 0.90	28 0.99
	29 1.04	30 0.99	31 1.16	32 0.94	33 1.22	34 0.88	35 1.22	36 0.89
45 0.80	37 1.01	38 1.13	39 0.88	40 1.11	41 0.88	42 0.92	43 0.96	44 1.00
54 1.01	46 0.87	47 0.92	48 1.24	49 0.90	50 1.22	51 0.97	52 1.18	53 0.97
	55 1.17	56 0.91	57 1.02	58 0.99	59 0.89	60 1.00	61 0.97	62 0.86

NOTE: X = MAXIMUM 1-PIN PEAK = 1.44

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 2 CYCLE 7 ASSEMBLY RELATIVE POWER DENSITY AT EOC EQUILIBRIUM XENON	Figure 5-3

CEA BANK 5 LOCATION

							1 0.66	2 0.86
				3 0.89	4 1.12	5 0.96	6 0.67	7 0.90
			8 0.96	9 1.30	10 1.07	11 1.13	12 0.73	13 0.48
		14 0.96	15 1.06	16 1.15	17 1.30	18 0.85	19 1.19	20 0.88
	21 0.89	22 1.30	23 1.15	24 1.04	25 1.05	26 1.16	27 0.86	28 0.96
	29 1.10	30 1.07	31 1.29	32 1.02	33 1.36 X	34 0.89	35 1.30	36 0.90
	37 0.95	38 1.12	39 0.85	40 1.15	41 0.89	42 0.95	43 1.01	44 1.06
45 0.65	46 0.68	47 0.73	48 1.19	49 0.86	50 1.32	51 1.04	52 1.29	53 1.03
54 0.86	55 0.90	56 0.48	57 0.88	58 0.96	59 0.90	60 1.06	61 1.03	62 0.80

NOTE: X = MAXIMUM 1-PIN PEAK = 1.59

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 2 CYCLE 7 ASSEMBLY RELATIVE POWER DENSITY WITH BANK 5 INSERTED, HFP, BOC	Figure 5-4

CEA BANK 5 LOCATION

						1 0.71	2 0.87	
				3 0.88	4 1.06	5 0.95	6 0.72	7 0.89
			8 0.96	9 1.20	10 1.03	11 1.09	12 0.74	13 0.48
		14 0.96	15 1.04	16 1.10	17 1.23	18 0.88	19 1.13	20 0.85
	21 0.88	22 1.20	23 1.10	24 1.03	25 1.03	26 1.15	27 0.90	28 0.98
	29 1.06	30 1.03	31 1.23	32 1.02	33 1.32	34 0.94	35 1.30	36 0.95
45 0.72	37 0.96	38 1.09	39 0.88	40 1.15	41 0.94	42 1.01	43 1.05	44 1.10
54 0.87	46 0.73	47 0.75	48 1.13	49 0.90	50 1.31	51 1.06	52 1.30 X	53 1.07
	55 0.89	56 0.48	57 0.85	58 0.98	59 0.95	60 1.10	61 1.07	62 0.86

NOTE: X = MAXIMUM 1-PIN PEAK = 1.47

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CALVERT CLIFFS UNIT 2 CYCLE 7 ASSEMBLY RELATIVE POWER DENSITY WITH BANK 5 INSERTED, HFP, EOC	Figure 5-5

6.0 THERMAL HYDRAULIC DESIGN

6.1 DNBR Analysis

Steady state DNBR analyses of Cycle 7 at the rated power level of 2700 MWt have been performed using the TORC computer code described in Reference 1, the CE-1 critical heat flux correlation described in Reference 2, and the simplified modeling methods described in Reference 3.

A variant of TORC called CETOP, optimized for simplified modeling applications, was used in this cycle to develop the "design thermal margin model" described generically in Reference 3. Details of CETOP are discussed in Reference 4. CETOP was approved for use on Calvert Cliffs Units in Reference 5. CETOP is used only because it reduces computer costs significantly; no margin gain is expected or taken credit for.

Table 6-1 contains a list of pertinent thermal-hydraulic design parameters applicable to both safety analyses and the generation of reactor protective system setpoint information. The calculational factors (engineering heat flux factor, engineering factor on hot channel heat input, rod pitch and clad diameter factor) listed in Table 6-1 have been combined statistically with other uncertainty factors at the 95/95 confidence/probability level (Reference 6) to define a design limit on CE-1 minimum DNBR when iterating on power as discussed in Reference 6 and approved by the NRC in Reference 5.

The statistically derived DNBR limit is being reduced from the value of 1.23 to a value of 1.21. The 1.23 value was approved in Reference 5. The reduction results from NRC approval in the Safety Evaluation Report (SER) of Reference 7 of a reduced CE-1 DNBR limit for CE's 14x14 fuel. At the time the Statistical Combination of Uncertainties (SCU) analysis was approved for the Calvert Cliffs units, NRC review of Reference 8 which dealt with the applicability of the CE-1 CHF correlation to rods with nonuniform Axial Power Distributions (APD) was incomplete. An interim CE-1 DNBR limit of 1.19 was thus used in the original SCU analysis. In the SER (Reference 7) on CE's nonuniform APD topical report, the NRC reduced the CE-1 DNBR limit from 1.19 to 1.15 for 14x14 fuel. The SCU DNBR limit is being correspondingly reduced from 1.23 to 1.21. The 1.21 SCU DNBR limit includes the following penalties imposed by the NRC in their review of the SCU analysis (Reference 5):

- Critical Heat Flux (CHF) correlation cross validation penalty (5% increase in standard deviation of CHF correlation uncertainty distribution)
- T-H code uncertainty penalty (5%, equal to two standard deviations).

The 1.21 SCU DNBR limit also includes a 0.006 DNBR rod bow penalty which accounts for the adverse effects of rod bowing on CHF for 14x14 fuel with burnup not exceeding 45 GWD/T.

The axial fuel densification factor is being reduced from 1.01 to 1.002 to make it consistent with existing calculations. The value of this parameter is being changed in the Unit 2 Tech. Specs via this submittal (see Section 9). This change was made in other safety analyses for the reference cycle and an identical change was made in the Unit 1 Tech. Specs. via the reference cycle submittal (Reference 9).

6.2 Effects of Fuel Bowing on DNBR Margin

The effects of fuel rod bowing on DNB margin for Calvert Cliffs Unit 2 Cycle 7 have been evaluated using the methods described in Reference 10. These methods were approved by NRC in Reference 11.

Based upon these methods, a penalty of 0.006 DNBR units is required to account for the adverse T-H effects of rod bow at an assembly average burnup of 45 GWD/T. This penalty was included in the derivation of the newly developed DNBR limit discussed above. For Unit 2 Cycle 7 no assembly is projected, based upon the burnup ranges discussed in previous sections, to have an average burnup greater than 45 GWD/T (see Figure 3-4).

TABLE 6-1

CALVERT CLIFFS UNIT 1
THERMAL-HYDRAULIC PARAMETERS AT FULL POWER**

General Characteristics	Unit	Reference+ Unit 1, Cycle 8	Unit 2, Cycle 7
Total Heat Output (core only)	MWt 10^6 BTU/hr	2700 9215	2700 9215
Fraction of Heat Generated In Fuel Rod		.975	.975
Primary System Pressure (Nominal)	psia	2250	2250
Inlet Temperature	^oF	548	548
Total Reactor Coolant Flow (steady state)	gpm 10^6 lb/hr	381,600 143.8	381,600 143.8
Coolant Flow Through Core	10^6 lb/hr	138.5	138.5
Hydraulic Diameter (nominal channel)	ft	0.044	0.044
Average Mass Velocity	10^6 lb/hr-ft^2	2.59	2.59
Pressure Drop Across Core (steady state flow irreversible P over entire fuel assembly)	psi	11.1	11.1
Total Pressure Drop Across Vessel (based on steady state flow and nominal dimensions)	psi	34.7	34.7
Core Average Heat Flux (Accounts for above fraction of heat generated in fuel rod and axial densification factor)	BTU/hr-ft^2	182,300***	180,700****
Total Heat Transfer Area (Accounts for axial densification factor)	ft^2	49,300***	49,700****
Film Coefficient at Average Conditions	BTU/hr-ft^2-^oF	5930	5930

TABLE 6-1
(continued)

General Characteristics	Unit	Reference+ Unit 1, Cycle 8	Unit 2, Cycle 7
Average Film Temperature Difference	°F	31	31
Average Linear Heat Rate of Undensified Fuel Rod (accounts for above fraction of heat generated in fuel rod)	kw/ft	6.09***	6.09****
Average Core Enthalpy Rise	BTU/lb	66.5	66.5
Maximum Clad Surface Temperature	°F	657	657

Calculational Factors	Reference+ Unit 1, Cycle 8	Unit 2, Cycle 7
Engineering Heat Flux on Hot Channel	1.03*	1.03*
Engineering Factor on Hot Channel Heat Input	1.02*	1.02*
Rod Pitch and Clad Diameter Factor	1.065*	1.065*
Fuel Densification Factor (axial)	1.01++	1.002+++

Notes

*These factors have been combined statistically with other uncertainty factors at 95/95 confidence/probability level (Reference 6) to define a design limit on CE-1 minimum DNBR when iterating on power as discussed in Reference 6 and approved by the NRC in Reference 5. This limit was verified to be applicable to Cycle 7.

**Due to the statistical combination of uncertainties described in References 6, 12, and 13, the nominal inlet temperature and nominal primary system pressure were used to calculate some of these parameters.

***Based on a value of 256 shims and 5 non-fuel rods.

****Based on a value of 224 shims.

+Reference cycle (Unit 1, Cycle 8) analysis is contained in Reference 3.

++This value was conservative with respect to existing calculations.

+++This value is consistent with existing calculations.

7.0 TRANSIENT ANALYSIS

The Design Basis Events (DBEs) considered in the Unit 2 Cycle 7 safety analyses are listed in Table 7-1. Core parameters input to the safety analyses for evaluating approaches to DNB and centerline temperature to melt fuel design limits are presented in Table 7-2.

As indicated in Table 7-1, no reanalysis was performed for any of the DBEs since key transient input parameters all lie within the bounds (conservative with respect to) of the reference cycle values (Unit 1, Cycle 8, Reference 1). The results and conclusions quoted in the reference cycle analysis for all DBEs are valid for Unit 2, Cycle 7.

TABLE 7-1

CALVERT CLIFFS UNIT 2, CYCLE 7
DESIGN BASIS EVENTS CONSIDERED IN THE NON-LOCA SAFETY ANALYSIS

Event	Analysis Status
7.1 Anticipated Operational Occurrences for which intervention of the RPS is necessary to prevent exceeding acceptable limits:	
7.1.1 Boron Dilution	Not Reanalyzed
7.1.2 Startup of an Inactive Reactor Coolant Pump[1]	Not Reanalyzed
7.1.3 Loss of Load	Not Reanalyzed
7.1.4 Excess Load	Not Reanalyzed
7.1.5 Loss of Feedwater Flow	Not Reanalyzed
7.1.6 Excess Heat Removal due to Feedwater Malfunction	Not Reanalyzed
7.1.7 Reactor Coolant System Depressurization	Not Reanalyzed
7.1.8 Excessive Charging Event	Not Reanalyzed
7.2 Anticipated Operational Occurrences for which RPS trips and/or sufficient initial steady state thermal margin, maintained by the LCOs, are necessary to prevent exceeding the acceptable limits:	
7.2.1 Sequential CEA Group Withdrawal[2]	Not Reanalyzed
7.2.2 Loss of Coolant Flow[3]	Not Reanalyzed
7.2.3 Full Length CEA Drop	Not Reanalyzed
7.2.4 Transients Resulting from the Malfunction of One Steam Generator[4]	Not Reanalyzed
7.2.5 Loss of AC Power[3]	Not Reanalyzed
7.3 Postulated Accidents	
7.3.1 CEA Ejection	Not Reanalyzed
7.3.2 Steam Line Rupture	Not Reanalyzed
7.3.3 Steam Generator Tube Rupture	Not Reanalyzed
7.3.4 Seized Rotor[3]	Not Reanalyzed

[1]Technical Specifications preclude this event during operation.

[2]Requires High Power and Variable High Power Trip.

[3]Requires Low Flow Trip.

[4]Requires trip on high differential steam generator pressure.

TABLE 7-2

CALVERT CLIFFS UNIT 2, CYCLE 7
CORE PARAMETERS INPUT TO SAFETY ANALYSES
FOR DNB AND CTM (CENTERLINE TO MELT) DESIGN LIMITS

Physics Parameters	Units	Reference Cycle** (Unit 1, Cycle 8)	Unit 2, Cycle 7
Radial Peaking Factors			
For DNB Margin Analyses (F_r^T)			
Unrodded Region		1.70*,+	1.70*,+
Bank 5 Inserted		1.87*,+	1.87*,+
For Planar Radial Component (F_{xy}^T) of 3-D Peak (CTM Limit Analyses)			
Unrodded Region		1.70*	1.70*
Bank 5 Inserted		1.87*	1.87*
Maximum Augmentation++ Factor		1.0	1.0
Moderator Temperature Coefficient	$10^{-4}\Delta\rho/^{o}F$	-2.7 → +.7	-2.7 → +.7
Shutdown Margin (Value Assumed in Limiting EOC Zero Power SLB)	$\%\Delta\rho$	-3.5	-3.5
Tilt Allowance	%	3.0	3.0

*For DNBR and CTM calculations, effects of uncertainties on these parameters were accounted for statistically. The procedures used in the Statistical Combination of Uncertainties (SCU) as they pertain to DNB and CTM limits are detailed in References 2, 3 and 4. These procedures have been approved by NRC for the Calvert Cliffs Units in Reference 5.

**Reference 1

+The values assumed are conservative with respect to the Technical Specification limits.

++Since the need for augmentation factors has been effectively removed as an issue (see Section 5.1.3.3), this item will be eliminated in future license submittals.

TABLE 7-2
(continued)

Safety Parameters	Units	Reference Cycle (Unit 1, Cycle 8)	Unit 2, Cycle 7
Power Level	MWt	2700*	2700*
Maximum Steady State Temperature	°F	548*	548*
Minimum Steady State RCS Pressure	psia	2200*	2200*
Reactor Coolant Flow	10^6 lbm/hr	138.5*	138.5*
Negative Axial Shape LCO Extreme Assumed at Full Power (Ex-Cores)	I_p	-.15*,+	-.15*,+
Maximum CEA Insertion at Full Power	% Insertion of Bank 5	25	25
Maximum Initial Linear Heat Rate for Transient Other than LOCA	KW/ft	16.0	16.0
Steady State Linear Heat Rate for Fuel CTM Assumed in the Safety Analysis	KW/ft	22.0	22.0
CEA Drop Time from Removal of Power to Holding Coils to 90% Insertion	sec	3.1	3.1
Minimum DNBR (CE-1)		1.23*	1.21*

*For DNBR and CTM calculations, effects of uncertainties on these parameters were accounted for statistically. The procedures used in the Statistical Combination of Uncertainties (SCU) as they pertain to DNB and CTM limits are detailed in References 2, 3 and 4. These procedures have been approved by NRC for the Calvert Cliffs Units in Reference 5.

+The values assumed are conservative with respect to the Technical Specification limits.

8.0 ECCS ANALYSIS

8.1 Large Break Loss-of-Coolant Accident

8.1.1 Introduction and Summary

An ECCS performance analysis was performed for Calvert Cliffs Unit 2 Cycle 7 to demonstrate compliance with 10CFR50.46 which presents the NRC Acceptance Criteria for Emergency Core Cooling Systems for Light Water Reactors (Reference 1). The analysis justifies an allowable Peak Linear Heat Generation Rate (PLHGR) of 15.5 kw/ft. This PLHGR is equal to the existing limit for Calvert Cliffs Units 1 and 2. The method of analysis and detailed results which support this value are presented in the following sections.

8.1.2 Method of Analysis

The ECCS performance analysis for Calvert Cliffs Unit 2 Cycle 7 consisted of an evaluation of the differences between Cycle 7 and the reference cycle, Unit 1 Cycle 8. Acceptable ECCS performance was demonstrated for the reference cycle in Reference 2 and approved by the NRC in Reference 3. The NRC approved C-E large break evaluation model (Reference 4) was used to re-evaluate ECCS performance for the limiting large break LOCA. The blowdown hydraulic calculations, refill/reflood hydraulics calculations and steam cooling heat transfer coefficients of the reference cycle apply to Unit 2, Cycle 7 since there have been no significant changes to RCS hardware characteristics. Therefore, only the fuel rod conditions specific to Cycle 7 require a re-evaluation of the ECCS hot rod clad temperature and oxidation calculations. The NRC approved STRIKIN-II (Reference 5) code was used for this purpose. Hot rod augmentation factors were removed, as was done for the reference cycle.

Burnup dependent calculations were performed to determine the limiting conditions for the ECCS performance analysis. The nuclear and fuel thermal performance (FATES3 (Reference 6)) data used as input to the ECCS analysis considered high burnup effects specific to the Cycle 7 reload. Two STRIKIN-II temperature calculations were performed: one at a low rod average burnup which yields the maximum initial fuel stored energy and one at a high burnup which yields the highest initial rod pressure. The low burnup maximum fuel stored energy case was demonstrated to be limiting.

The temperature calculations for both cases were performed for the 1.0 Double Ended Slot at Pump Discharge (DES/PD) break. The break spectrum analysis performed for Unit 1 Cycle 2 (Reference 7) determined that the 1.0 DES/PD is the limiting break since it yields the highest residual fuel stored energy at the end of the blowdown period and, therefore, yields the highest peak cladding temperature during the reflood period.

8.1.3 Results

Table 8.1-1 summarizes the results calculated for the two rod average burnup cases selected. A summary of the fuel parameter input values is shown in Table 8.1-2. For comparison purposes, the corresponding values of the reference cycle analysis (Unit 1 Cycle 8) are also presented in Tables 8.1-1 and 8.1-2. A list of the significant parameters displayed graphically for the limiting case (Figures 8.1-1 through 8.1-6) is presented in Table 8.1-3.

The results of the evaluation confirm that 15.5 kw/ft is an acceptable value for the PLHGR in Cycle 7. As shown in Table 8.1-1, the peak clad temperature, maximum local oxidation and core wide clad oxidation values of 1945^{o}F, 5.4% and <.51%, respectively, are well below the 10CFR50.46 acceptance criteria limits of 2200^{o}F, 17% and 1%, respectively. These results have been confirmed for up to 100 plugged tubes per steam generator.

As shown in Table 8.1-1, the burnup with the maximum initial stored energy in the fuel (943 MWD/MTU) resulted in the highest peak clad temperature of 1945^{o}F. The transient results for the limiting case are presented in Figures 8.1-1 to 8.1-6. The fuel cladding is predicted to rupture, as well as achieve its peak temperature, during the reflood period (at 33 seconds and 261 seconds, respectively).

The high burnup (50,100 MWD/MTU) case resulted in a peak clad temperature of 1929^{o}F, 16^{o}F lower than that for the maximum initial fuel stored energy case. This case assumed that the rod operates at a PLHGR of 15.5 kw/ft. This assumption is very conservative since a rod at such a high burnup would actually be at a power level significantly below 15.5 kw/ft. For the high burnup case, at 15.5 kw/ft the pin pressure was sufficient to cause an earlier hot rod rupture time. However, the lower initial stored energy at that burnup causes a lower peak clad temperature than the case at 943 MWD/MTU, as shown in Table 8.1-1.

8.1.4 Conclusions

As discussed above, conformance to the ECCS criteria is summarized by the analysis results presented in Table 8.1-1. The most limiting case results in a peak clad temperature of 1945^{o}F, which is well below the acceptance limit of 2200^{o}F. The maximum local and core wide values for zirconium oxidation percentages, as shown in Table 8.1-1, remain well below the acceptance limit values of 17% and 1%, respectively. Therefore, operation of Unit 2 Cycle 7 at a PLHGR of 15.5 kw/ft and a power level of 2754 MWT (102% of 2700 MWT) is in compliance with the 10CFR50.46 acceptance criteria.

Reference 8 notified the NRC of a potential non-conservatism in one element of the C-E large break LOCA evaluation model (Reference 4). As stated in Reference 8, the effect of this non-conservatism on peak clad temperature (PCT) is expected to be well within the 255^{o} margin to the 2200^{o}F PCT limit which the current analysis demonstrates. Consequently, it has been concluded that this potential non-conservatism will not alter the previously discussed determination that the operation of Unit 2 Cycle 7 would be in compliance with the 10CFR50.46 acceptance criteria.

TABLE 8.1-1

Summary of ECCS Performance Results for
Calvert Cliffs 2, Cycle 7 for the Limiting Break Size
(1.0 DES/PD) as Compared to Reference Cycle (Unit 1, Cycle 8)

Parameters	Limiting Case (Maximum Initial Fuel Stored Energy)		High Burnup Case (Maximum Initial Rod Pressure)	
	Unit 1 Cycle 8	Unit 2 Cycle 7	Unit 1 Cycle 8	Unit 2 Cycle 7
Rod Average Burnup MWD/MTU	943	943	50,000	50,100
Peak Clad Temperature (PCT), °F	1836+	1945	1822+	1929
Time of PCT, Seconds	254+	261	254+	263
Time of Clad Rupture, Seconds	35.0+	33.0	10.5+	10.7
Peak Clad Oxidation, %	3.70+	5.40	3.75+	5.40
Core Wide Oxidation, %	<.51	<.51	<.51	<.51

+These are revised values; revision of these values was made soon after the submission of Reference 2 and did not affect the conclusions contained in Section 8.1 of Reference 2.

TABLE 8.1-2

Calvert Cliffs Unit 2 Cycle 7 Fuel Parameters
as Compared to Unit 1 Cycle 8

Fuel Parameters	Unit 1 Cycle 8 Values	Unit 2 Cycle 7 Values
Reactor Power Level (102% of Nominal) MWT	2754	2754
Average Linear Heat Rate (102% of Nominal) kw/ft	6.37	6.37
Hot Channel Peak Linear Heat Generation Rate kw/ft	15.5	15.5
Hot Assembly Peak Linear Heat Generation Rate kw/ft	12.52	13.42
*Gap Conductance at PLHGR (B/hr-ft^2-^oF)	2057$^+$	1937
*Fuel Centerline Temperature at PLHGR (^oF)	3655$^+$	3649
*Fuel Average Temperature at PLHGR (^oF)	2221$^+$	2228
*Hot Rod Gas Pressure (psia)	1186$^+$	1188
*Hot Rod Burnup (MWD/MTU)	943	943
Hot Rod Augmentation Factor (Maximum)	1.00	1.00

*Are initial fuel rod parameters, in STRIKIN-II, which yield maximum PCT.

$^+$These are revised values; revision of these values was made soon after the submission of Reference 2 and did not affect the conclusions contained in Section 8.1 of Reference 2.

TABLE 8.1-3

Calvert Cliffs Unit 2 Cycle 7
Reload Analysis Plots for Limiting Case

Variable	Figure Number
Peak Clad Temperature	8.1-1
Hot Spot Gap Condutance	8.1-2
Peak Local Clad Oxidation	8.1-3
Temperature of Fuel Centerline, Fuel Average, Clad and Coolant at Hottest Node	8.1-4
Hot Spot Heat Transfer Coefficient	8.1-5
Hot Rod Internal Gas Pressure	8.1-6

FIGURE 8.1-1
CALVERT CLIFFS 2 CYCLE 7
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
PEAK CLAD TEMPERATURE

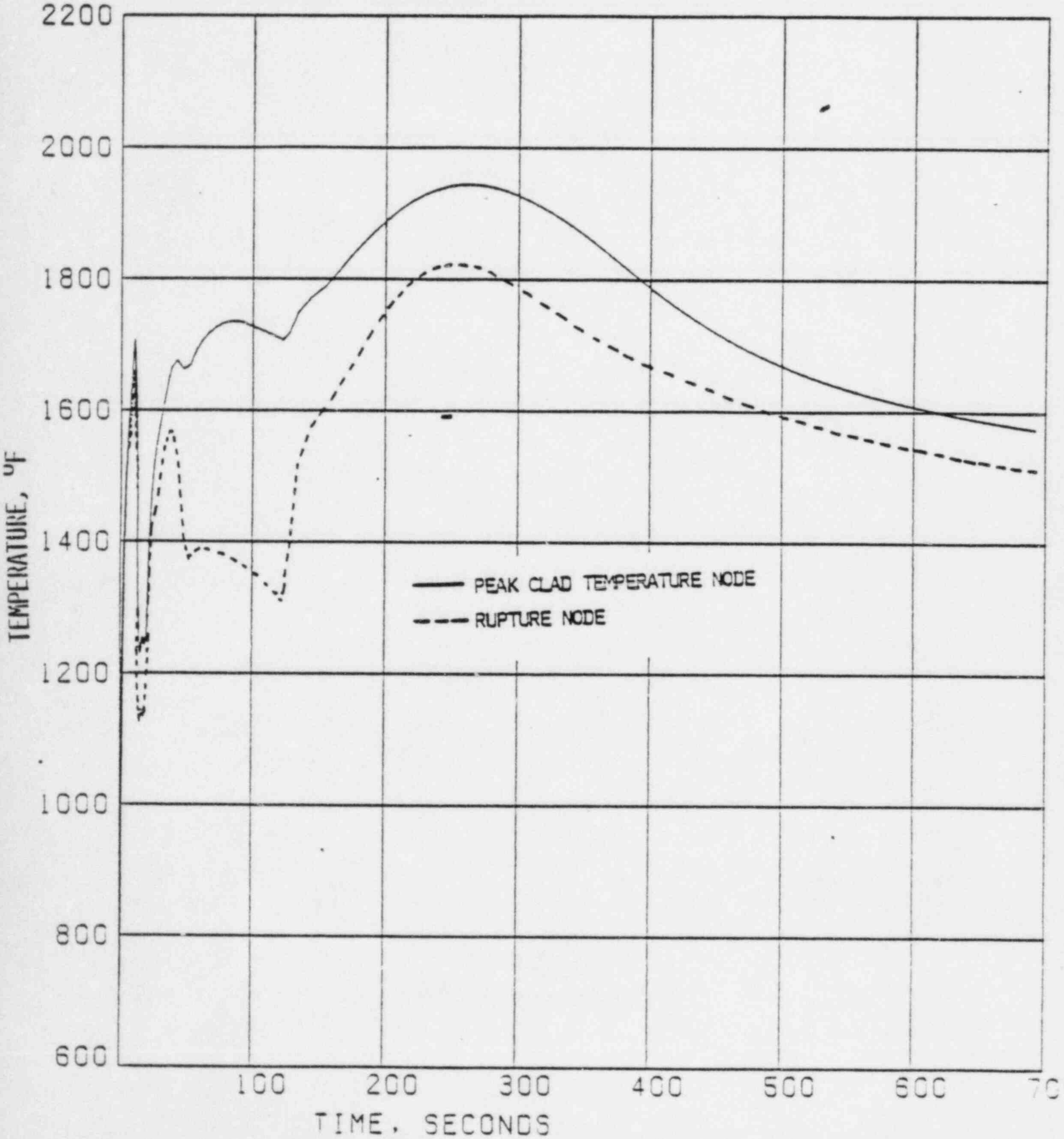


FIGURE 8.1-2
CALVERT CLIFFS 2 CYCLE 7
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
HOT SPOT GAP CONDUCTANCE

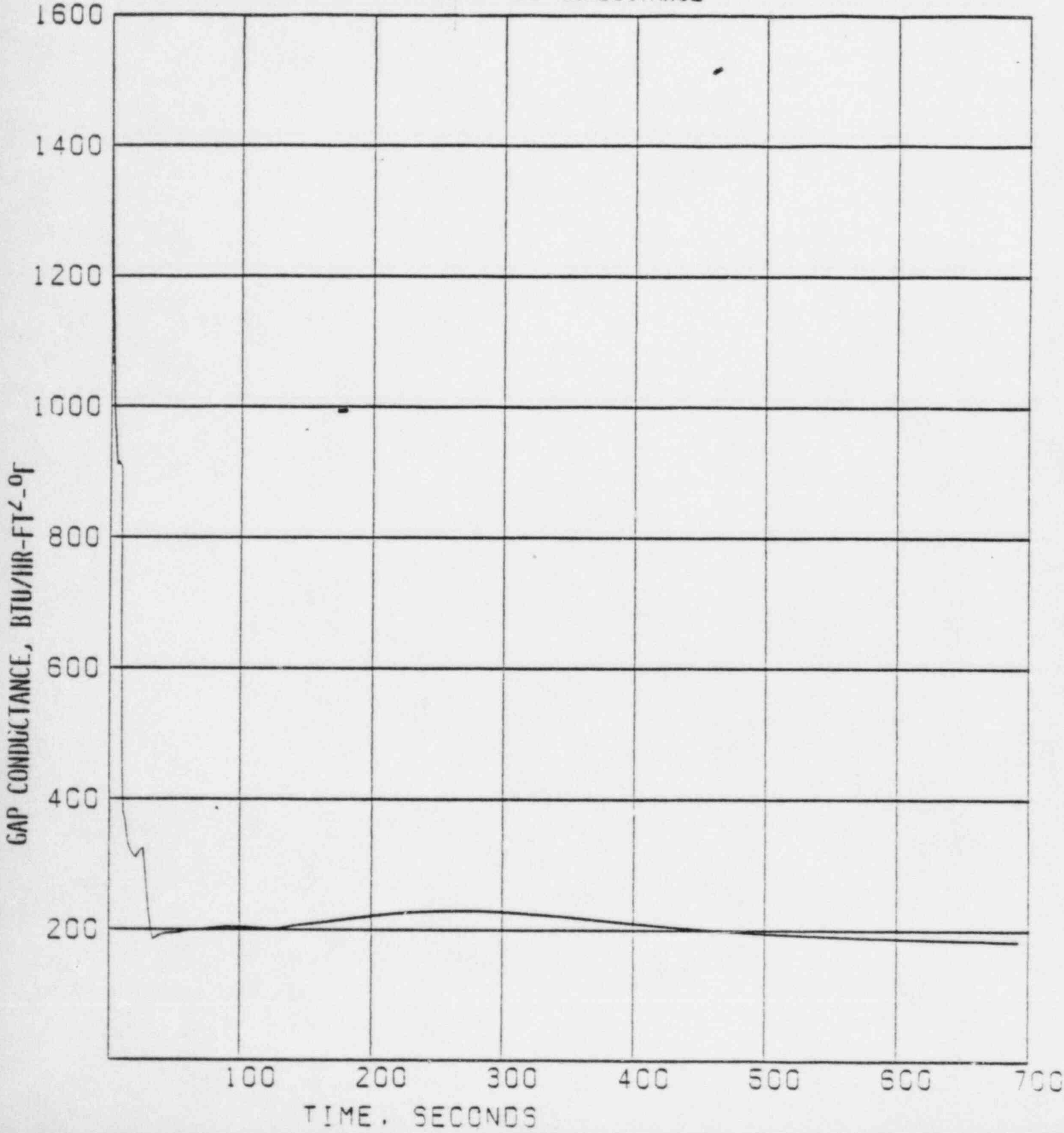


FIGURE 8.1-3
CALVERT CLIFFS 2 CYCLE 7
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
PEAK LOCAL CLAD OXIDATION

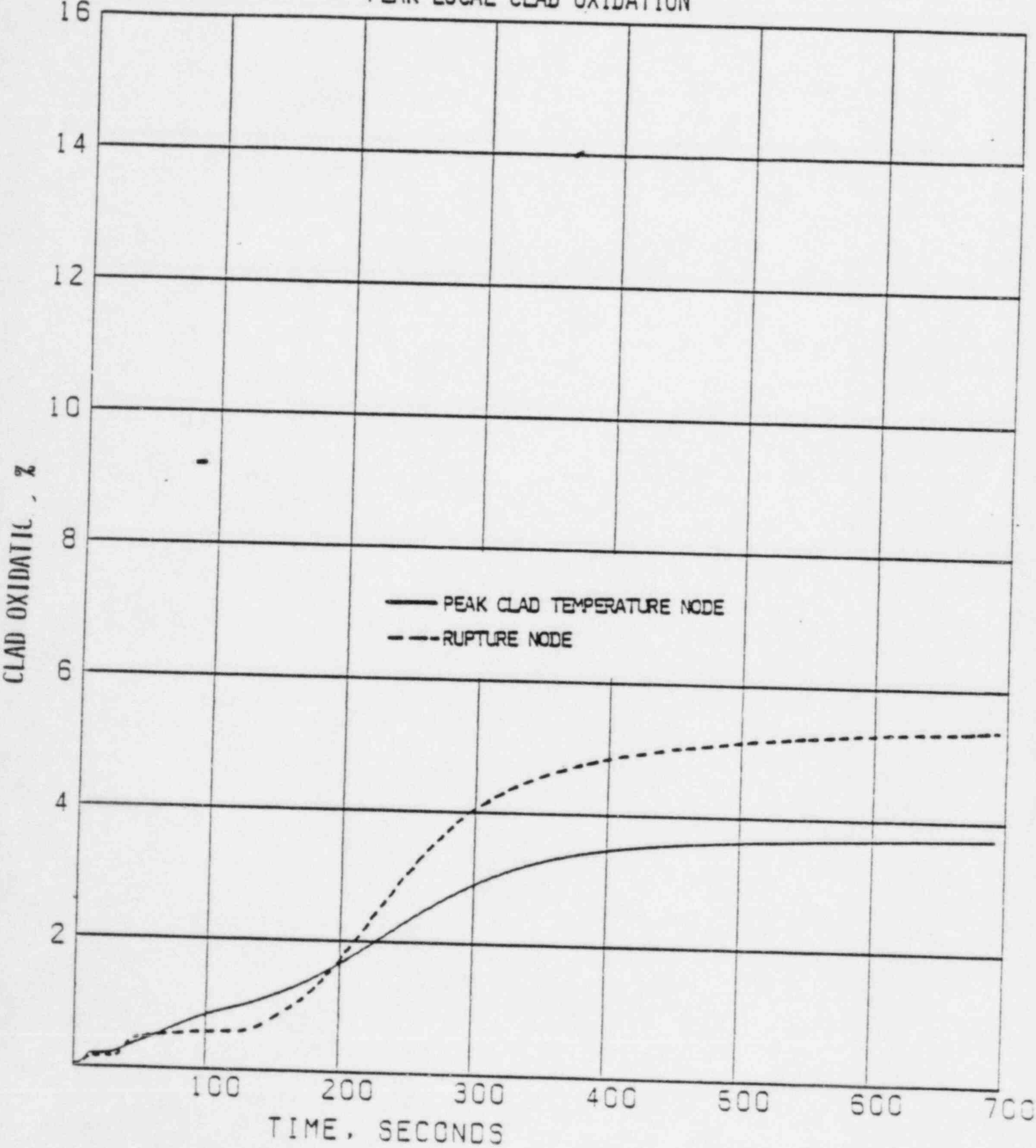


FIGURE 8.1-4
CALVERT CLIFFS 2 CYCLE 7
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
TEMPERATURE OF FUEL CENTERLINE, AVERAGE FUEL,
CLAD AND COOLANT OF HOTTEST NODE

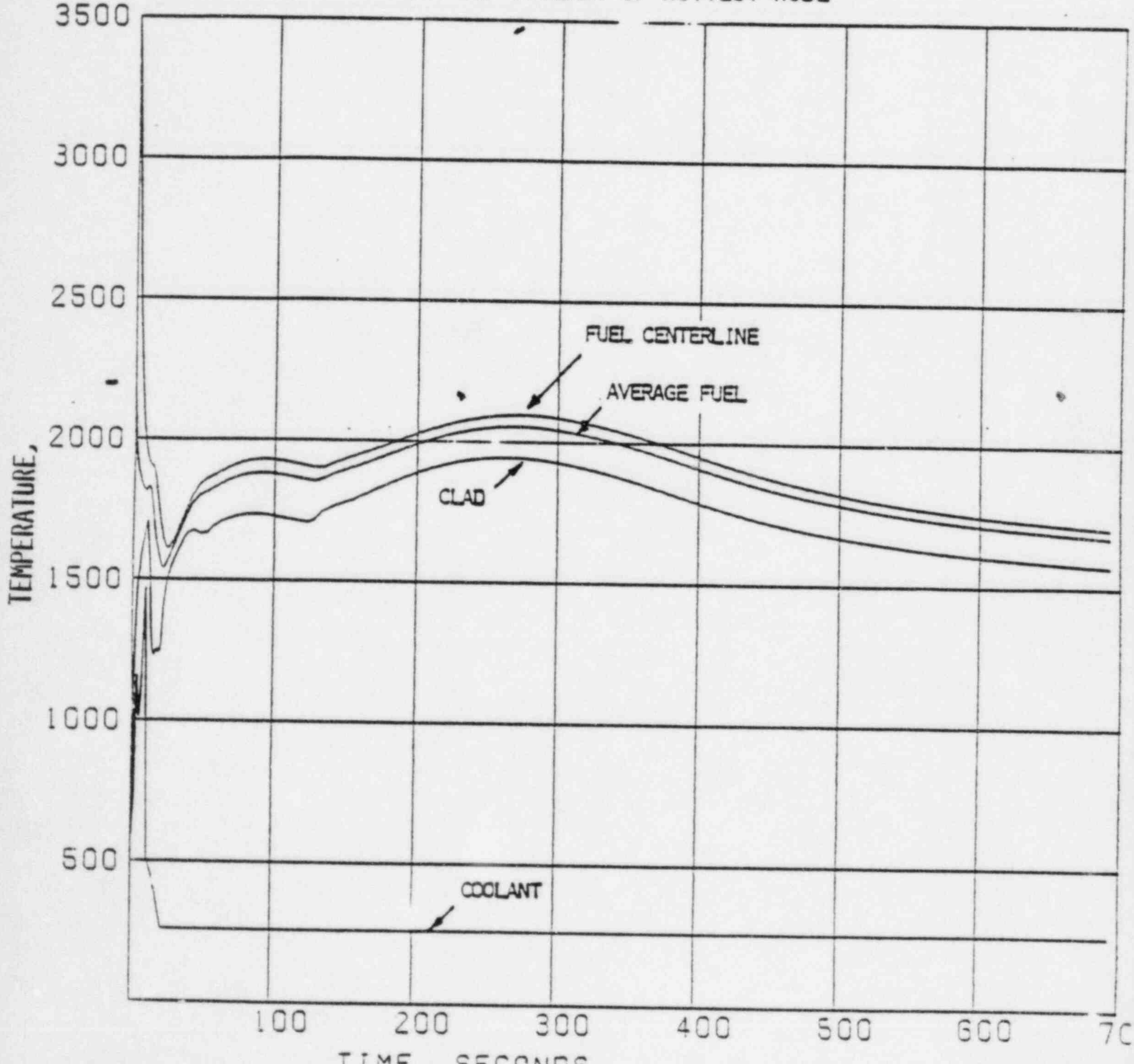


FIGURE 8.1-5
CALVERT CLIFFS 2 CYCLE 7
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE
HOT SPOT HEAT TRANSFER COEFFICIENT

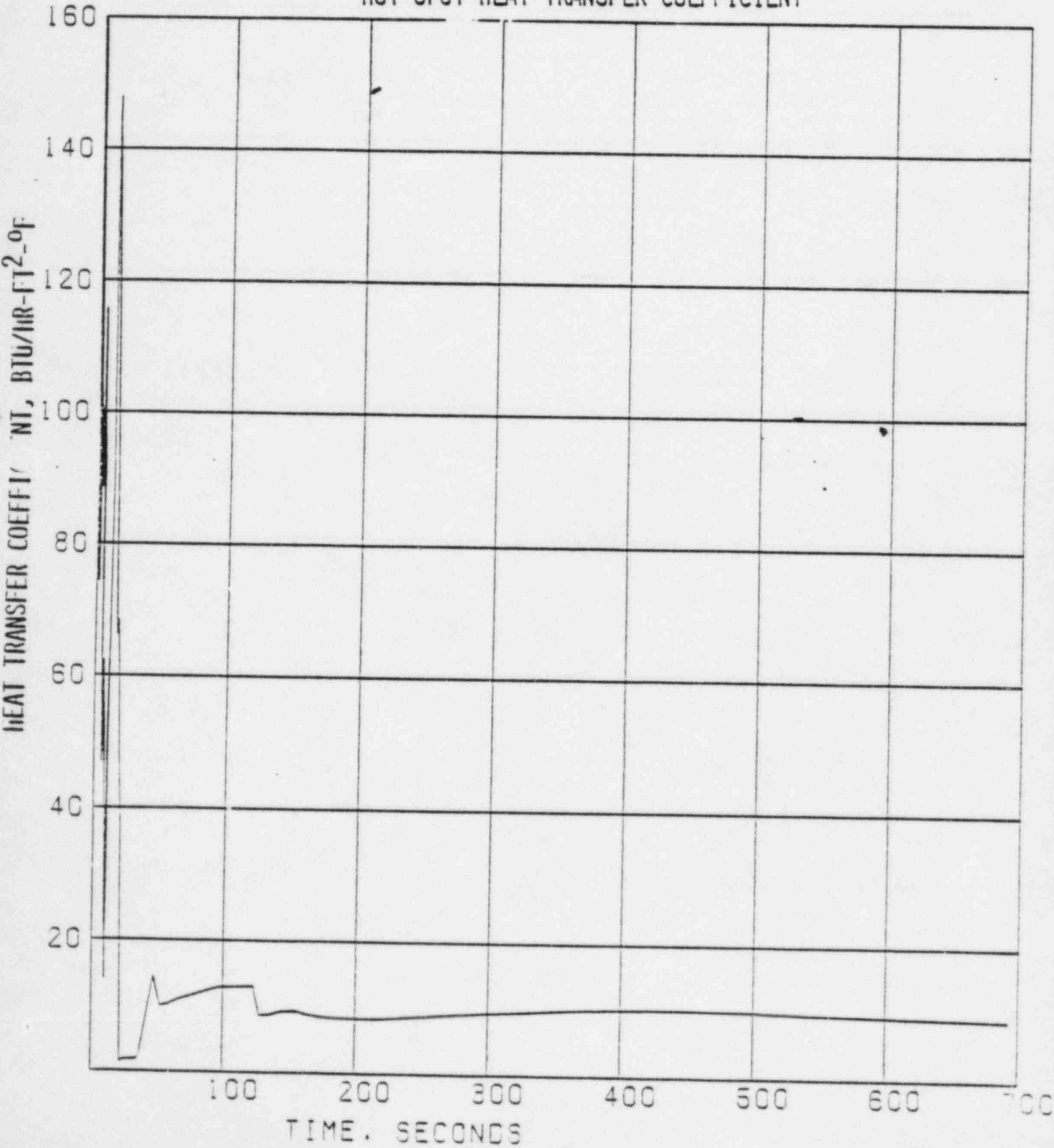
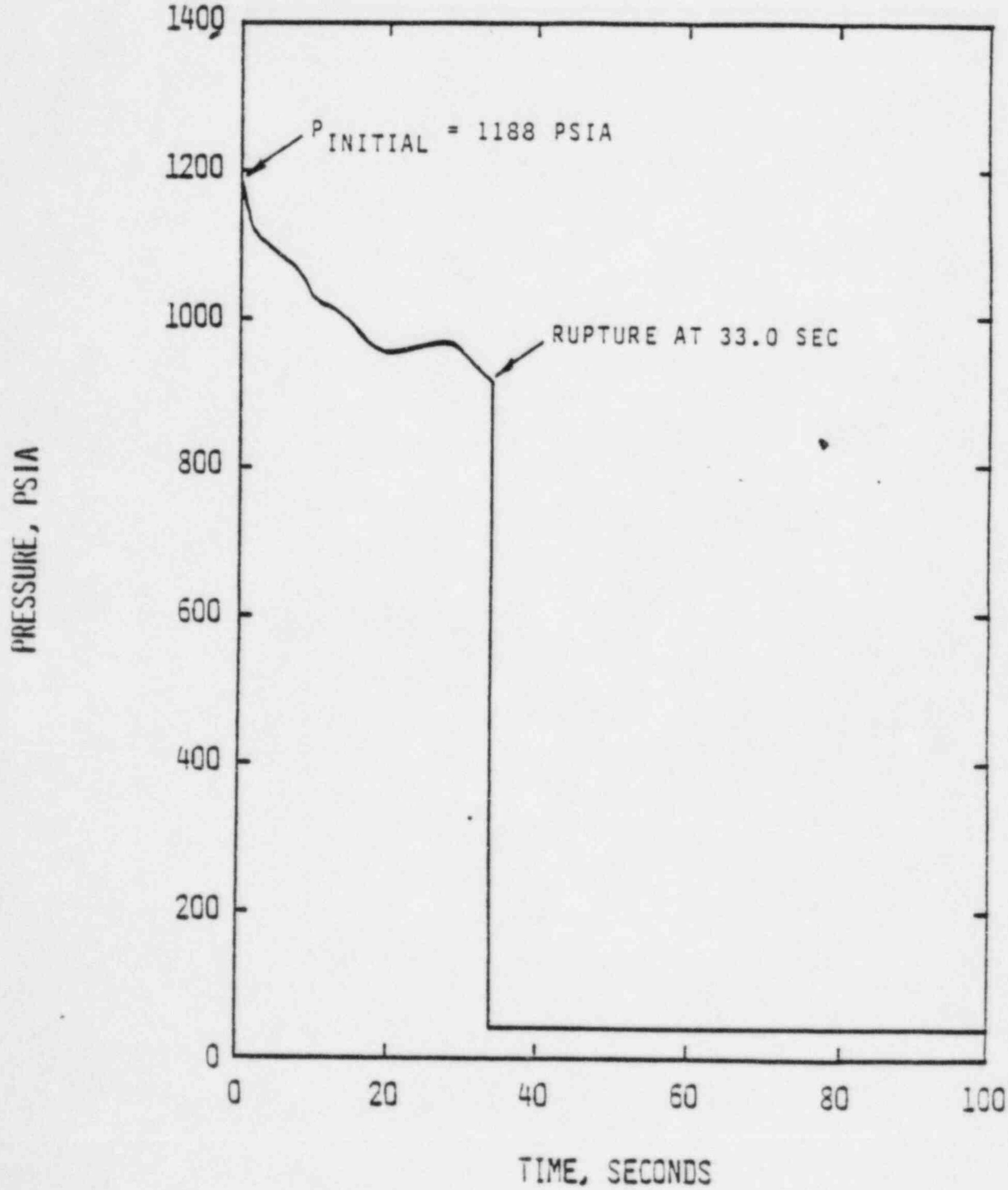


FIGURE 8.1-6
CALVERT CLIFFS UNIT 2 CYCLE 7
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
HOT ROD INTERNAL GAS PRESSURE



8.2 Small Break Loss-of-Coolant Accident

Analyses have confirmed that the reported small break loss-of-coolant accident results (Reference 2) of the reference cycle - Unit 1 Cycle 8 - bound Calvert Cliffs Unit 2 Cycle 7. These results have been approved by the NRC in Reference 3. Therefore, acceptable small break LOCA ECCS performance is demonstrated at a peak linear heat rate of 15.5 kw/ft and a reactor power level of 2754 MWt (102% of 2700 MWt). This acceptable performance has been confirmed with up to 100 plugged tubes per steam generator.