9.0 TECHNICAL SPECIFICATIONS

The Technical Specification changes which are being requested in order to make the Calvert Cliffs Unit 2 Technical Specifications consistent with either the reference cycle (Reference 1) analyses which have been verified for Unit 2 Cycle 7 or the analyses contained herein are presented in this section. All changes except two which are being requested herein for Unit 2 were approved for Unit 1 in Reference 2. The first new change makes the affected Unit 2 Technical Specification identical to the corresponding Unit 1 Technical Specification. The second new change lowers the minimum DNBR Technical Specification limit to make it consistent with the final DNBR limit approved in Reference 3.

Table 9-1 presents a summary of the Technical Specification changes, in the form of: 1) an action statement for each change; 2) the reason for each change and 3) a reference to the supporting analyses which demonstrate acceptable safety analyses results for each change. Following Table 9-1 the existing Technical Specification page with the intended modification is provided for each Technical Specification for which a change is being requested.

The Technical Specification changes requested herein for Unit 2 are identical to those that were approved for Unit 1 in Reference 2 with the following exceptions:

1) All HPSI flow related changes for Unit 2 (including ASI limitations) have already been submitted in Reference 4 for application to Cycle 6.

2) The change in the surveillance interval for CEA insertability in Technical Specification 3/4.10.1 has already been submitted in Reference 5 for application to Cycle 6.

3) Changes in the lift setting values and format of the Main Steam Safety Valve (MSSV) Technical Specification, and changes to permit entry into Mode 3 with 2 MSSVs per steam generator operable are being submitted separately for application to Cycle 6.

4) The minimum DNBR SCU based limit (see Section 6) is being lowered to make it consistent with the final minimum non-SCU based DNBR limit approved by the NRC in Reference 3.

5) The radial peaking factor at which full power operation may proceed when operating on the excore monitoring system is being raised to make it identical to the Unit 1 value and to increase operating margin.

Table 9-1

Calvert Cliffs 2 Cycle 7
Technical Specification Changes

Tech. Spec. No. and Page	Action	Explanation	Support
B.2.1.1, B.2.2.1 pages B2-1, B2-3, B2-5, B2-6	Change minimum DNBR limit from 1.23 to 1.21	The minimum SCU based DNBR limit is being lowered to make it consistent with the final minimum non-SCU based DNBR limit approved by the NRC.	The discussion in Chapter 6, concerning the derivation of the new SCU based DNBR limit using the NRC approved non-SCU based final DNBR limit and previously approved SCU and rod bow penalty methodologies, supports this reduction.
3/4.1.1.1 page 3/4 1-1	Change shutdown margin, $T_{avg} > 200^{o}/F$, from 4.3% $\Delta k/k$ to 3.5% $\Delta k/k$	The shutdown margin is being lowered to accomodate the effects of extended burnup.	
3.1.1.4 page 3/4 1-5	Change MTC positive limit, Power $\leq$ 70%, from $+0.5 \times 10^{-4}$ $\Delta k/k/^{o}F$ to $+0.7 \times 10^{-4}$ $\Delta k/k/^{o}F$.	The MTC is being raised to accomodate the effects of long cycles and to simplify startup procedures.	The safety analyses presented in Chapter 7 of Reference 1, and verification in Chapter 7 of this document that these analyses are applicable to Unit 2 Cycle 7 support these changes.
	Change MTC negative limit from -2.5×10^{-4} $\Delta k/k/^{o}F$ to -2.7×10^{-4} $\Delta k/k/^{o}F$.	The MTC is being lowered to to accomodate the effects of of extended burnup.	
4.2.1.4 page 3/4 2-2	Remove flux peaking augmentation factors	Augmentation factors are being removed in recognition of the demonstrated lack of gap formation in pre-pressurized non-densifying fuel and to increase operating margin,	1) Detailed discussion and data was presented in Reference 6 to support this change. 2) The thermal design analysis of the fuel pins presented in Section 4.3 supports the change. 3) The ECCS performance analysis for the large break spectrum presented in Section 8.1 supports this change.

Table 9-1 (continued)

Tech. Spec. No. and Page	Action	Explanation	Support
4.2.1.4 (cont.)	Reduce the measurement-calculational uncertainty from 7.0% to 6.2%	This uncertainty is being reduced to conform to the approved value and to increase operating margin.	1) The new value is supported in Reference 7. 2) The thermal design analysis of the fuel pins presented in Section 4.3 supports this change.
	Reduce the axial fuel densification and thermal expansion factor from 1.0% to 0.2%	This uncertainty is being reduced to a level consistent with existing calculations and to increase operating margin.	The thermal design analysis of the fuel pins presented in Section 4.3 supports this change.
Figure 3.2-3b page 3/4 2-4a	Modify Figure 3.2-3b as indicated to increase the the radial peaking factor at which full power operation may proceed when operating on the excore monitoring system from 1.50 to 1.54	This radial peaking factor, in the form of the variable 'N', is being increased to make the Unit 2 Figure 3.2-3b identical to the corresponding Unit 1 figure and to increase operating margin.	The setpoint analysis for Unit 2 Cycle 7 supports this change.
Figure 4.2-1 page 3/4 2-5	Delete Figure 4.2-1	See change for Tech. Spec. 4.2.1.4 which covers removal of flux peaking augmentation factor.	See change for Tech. Spec. 4.2.1.4 which covers removal of flux peaking augmentation factors.

Table 9-1 (continued)

Tech. Spec. No. and Page	Action	Explanation	Support
B 3/4.1.1.1 and B 3/4.1.1.2 page B 3/4 1-1	Change EOC shutdown margin, T_{avg} >200^{0}F, from 4.3% $\Delta k/k$ to 3.5% $\Delta k/k$ and change BOC shutdown margin, T_{avg} >200^{0}F, from 4.3% $\Delta k/k$ to 3.5% $\Delta k/k$	See change for Tech. Spec. 3/4.1.1.1	See change for Tech. Spec. 3/4.1.1.1
B 3/4.2.1 page B 3/4 2-1	Remove flux peaking augmentation factors, change measurement-calculational uncertainty from 7.0% to 6.2% and change axial fuel densification and thermal expansion factor from 1.0% to 0.2%	See change for Tech. Spec. 4.2.1.4.	See change for Tech. Spec. 4.2.1.4.
B 3/4.2.5 page B 3/4 2-2	Change minimum DNBR limit from 1.23 to 1.21	See change for Tech. Specs. B.2.1.1 and B.2.2.1	See change for Tech. Specs. B.2.1.1 and B.2.2.1
	Insert the additional text concerning limiting criteria on the DNB LCO, as indicated	The BASES section for the DNB LCO is being expanded to more clearly define all of the criteria which are used to establish the Tech. Spec. values.	The text is merely updating the BASES to describe what has been standard practice.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate at or less than 22.0 kw/ft. Centerline fuel melting will not occur for this peak linear heat rate. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the CE-1 correlation. The CE-1 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ration of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.23 → 1.21. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figures 2.1-1, 2.1-2, 2.1-3 and 2.1-4 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and maximum cold leg temperature of various pump combinations for which the minimum DNBR is no less than 1.23 (1.21) for the family of axial shapes and corresponding radial peaks shown in Figure B2.1-1. The limits in Figures 2.1-1, 2.1-2, 2.1-3 and 2.1-4 were calculated for reactor coolant inlet temperatures less than or equal to 580°F. The dashed line at 580°F coolant inlet temperature is not a safety limit; however, operation above 580°F is not possible because of the actuation of the main steam line safety valves which limit the maximum value of reactor inlet temperature. Reactor operation at THERMAL POWER levels higher than 110% of RATED THERMAL POWER is prohibited by the high power level trip setpoint specified in

Table 2.1-1. The area of safe operation is below and to the left of these lines.

The conditions for the Thermal Margin Safety Limit curves in Figures 2.1-1, 2.1-2, 2.1-3 and 2.1-4 to be valid are shown on the figures.

The reactor protective system in combination with the Limiting Conditions for Operation, is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure, and THERMAL POWER level that would result in a DNBR of less than 1.23 and preclude the existence of flow instabilities.

1.21

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III, 1967 Edition, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.7, Class I, 1969 Edition, which permits a maximum transient pressure of 110% (2750 psia) of component design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

1.21

operation of the reactor at reduced power if one or two reactor coolant pumps are taken out of service. The low-flow trip setpoints and Allowable Values for the various reactor coolant pump combinations have been derived in consideration of instrument errors and response times of equipment involved to maintain the DNBR above 1.23 under normal operation and expected transients. For reactor operation with only two or three reactor coolant pumps operating, the Reactor Coolant Flow-Low trip setpoints, the Power Level-High trip setpoints, and the Thermal Margin/Low Pressure trip setpoints are automatically changed when the pump condition selector switch is manually set to the desired two- or three-pump position. Changing these trip setpoints during two and three pump 1.21 operation prevents the minimum value of DNBR from going below 1.23 during normal operational transients and anticipated transients when only two or three reactor coolant pumps are operating.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, backed up by the pressurizer code safety valves and main steam line safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is 100 psi below the nominal lift setting (2500 psia) of the pressurizer code safety valves and its concurrent operation with the power-operated relief valves avoids the undesirable operation of the pressurizer code safety valves.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated concurrently with a safety injection. The setpoint for this trip is identical to the safety injection setpoint.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setting of 685 psia is sufficiently below the full-load operating point of 850 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This setting was used with an uncertainty factor of ± 85 psi in the accident analyses which was based on the Main Steam Line Break event.

Steam Generator Water Level

The Steam Generator Water Level-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity and assures that the pressure of the reactor coolant system will not exceed its Safety Limit. The specified setpoint in combination with the auxiliary feedwater actuation system ensures that sufficient water inventory exists in both steam generators to remove decay heat following a loss of main feedwater flow event.

Axial Flux Offset

1.21

The axial flux offset trip is provided to ensure that excessive axial peaking will not cause fuel damage. The axial flux offset is determined from the axially split excore detectors. The trip setpoints ensure that neither a DNBR of less than 1.23 nor a peak linear heat rate which corresponds to the temperature for fuel centerline melting will exist as a consequence of axial power maldistributions. These trip setpoints were derived from an analysis of many axial power shapes with allowances for instrumentation inaccuracies and the uncertainty associated with the excore to incore axial flux offset relationship.

Thermal Margin/Low Pressure

The Thermal Margin/Low Pressure trip is provided to prevent operation when the DNBR is less than 1.23 1.21

The trip is initiated whenever the reactor coolant system pressure signal drops below either 1875 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power, reactor inlet temperature, and the number of reactor coolant pumps operating. The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} > 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be ≥ ~~4.3%~~ 3.5* Δk/k.

APPLICABILITY: MODES 1, 2**, 3 and 4.

ACTION:

With the SHUTDOWN MARGIN < ~~4.3%~~ 3.5* Δk/k, immediately initiate and continue boration at ≥ 40 gpm of 2300 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be ≥ ~~4.3%~~ 3.5* Δk/k:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. When in MODES 1 or 2#, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2##, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

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* Adherence to Technical Specification 3.1.3.6 as specified in Surveillance Requirements 4.1.1.1.1 assures that there is sufficient available shutdown margin to match the shutdown margin requirements of the safety analyses.

** See Special Test Exception 3.10.1.

\# With $K_{eff} \geq 1.0$.

\## With $K_{eff} < 1.0$.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

0.7

a. Less positive than 0.5 x 10^{-4} $\Delta k/k/°F$ whenever THERMAL POWER is ≤ 70% of RATED THERMAL POWER,

b. Less positive than 0.2 x 10^{-4} $\Delta k/k/°F$ whenever THERMAL POWER is > 70% of RATED THERMAL POWER, and

c. Less negative than -2.5 x 10^{-4} $\Delta k/k/°F$ at RATED THERMAL POWER.

2.7

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

*With K_{eff} ≥ 1.0.

#See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

c. Verifying at least once per 31 days that the AXIAL SHAPE INDEX is maintained within the limits of Figure 3.2-2, where 100 percent of the allowable power represents the maximum THERMAL POWER allowed by the following expression:

M x N

where:

1. M is the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

2. N is the maximum allowable fraction of RATED THERMAL POWER as determined by the F'_{xy} curve of Figure 3.2-3b.

4.2.1.4 Incore Detector Monitoring System - The incore detector monitoring system may be used for monitoring the core power distribution by verifying that the incore detector Local Power Density alarms:

a. Are adjusted to satisfy the requirements of the core power distribution map which shall be updated at least once per 31 days of accumulated operation in MODE 1.

b. Have their alarm setpoint adjusted to less than or equal to the limits shown on Figure 3.2-1 when the following factors are appropriately included in the setting of these alarms:

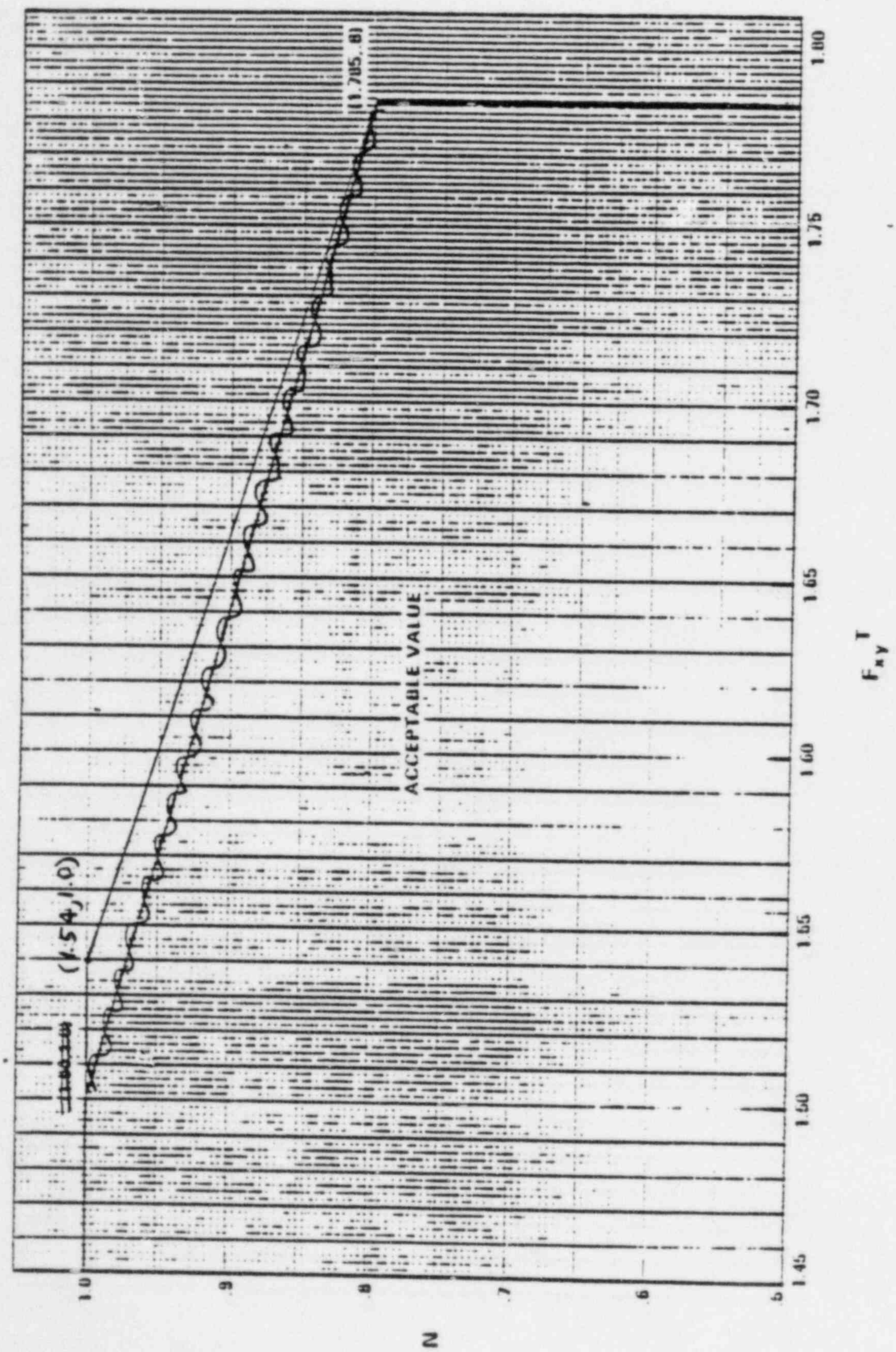
~~1. Flux peaking augmentation factors as shown in Figure 4.2-1,~~

2. 1. A measurement-calculational uncertainty factor of 1.07, 1.062

3. 2. An engineering uncertainty factor of 1.03,

4. 3. A linear heat rate uncertainty factor of 1.01 1.002 due to axial fuel densification and thermal expansion, and

5. 4. A THERMAL POWER measurement uncertainty factor of 1.02.



TOTAL PLANAR RADIAL PEAKING FACTOR vs N

Figure 3.2-3b

DELETE

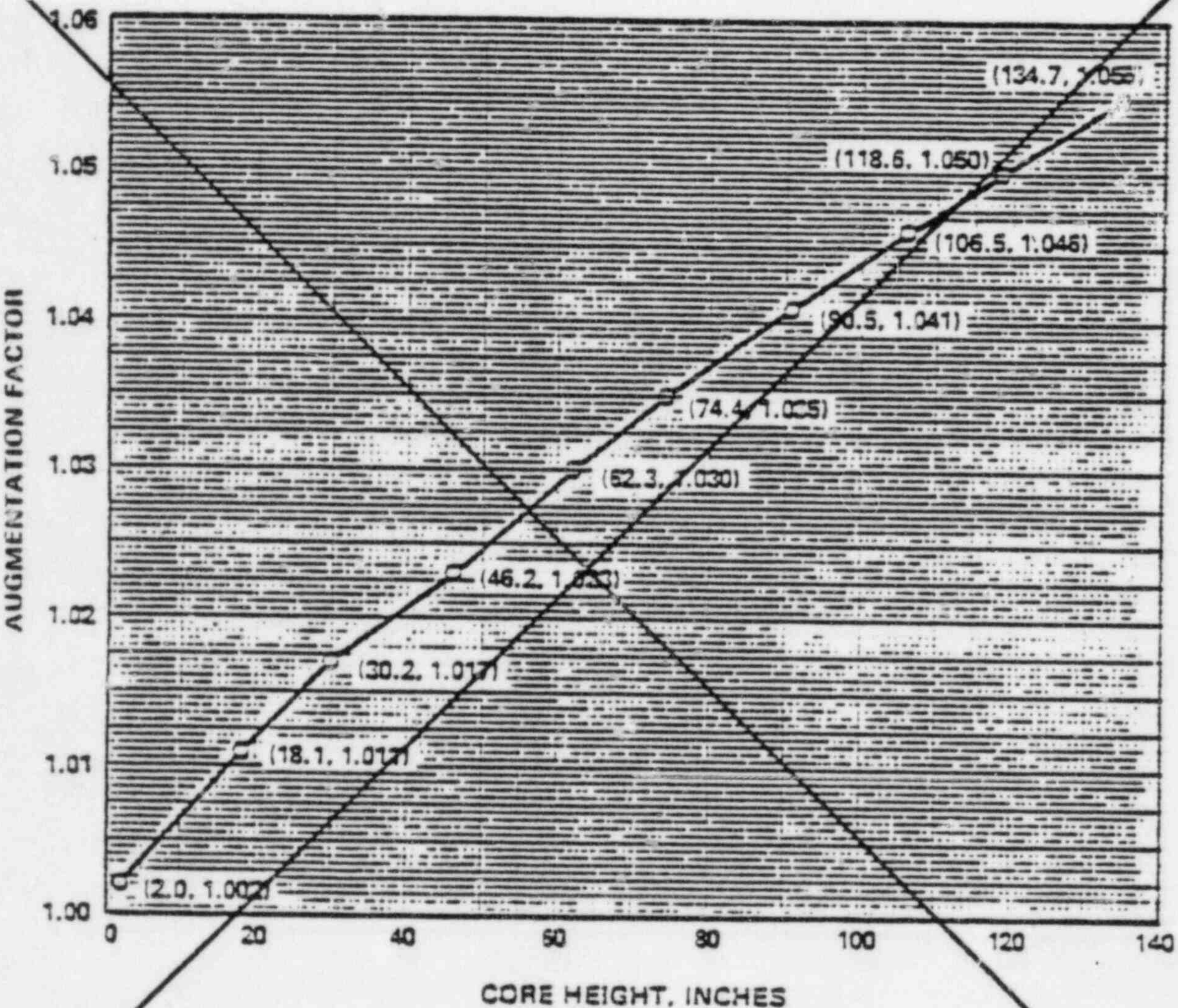


FIGURE 4.2-1
Augmentation Factor vs
Distance from Bottom of Core

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

3.5 3.5

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration and RCS T_{avg}. The minimum available SHUTDOWN MARGIN for no load operating conditions at beginning of life is 4.3% $\Delta k/k$ and at end of life is 4.3% $\Delta k/k$. The SHUTDOWN MARGIN is based on the safety analyses performed for a steam line rupture event initiated at no load conditions. The most restrictive steam line rupture event occurs at EOC conditions. For the steam line rupture event at beginning of cycle conditions, a minimum SHUTDOWN MARGIN of less than 4.3% $\Delta k/k$ is required to control the reactivity transient, and end of cycle conditions require 4.3% $\Delta k/k$. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With $T_{avg} \leq 200°F$, the reactivity transients resulting from any postulated accident are minimal and a 3% $\Delta k/k$ shutdown margin provides adequate protection. With the pressurizer level less than 90 inches, the sources of non-borated water are restricted to increase the time to criticality during a boron dilution event.

3.5 3.5

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 3000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 9,601 cubic feet in approximately 24 minutes. The reactivity change rate associated with boron concentration reductions will therefore be within the capability of operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

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3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System and the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the AXIAL SHAPE INDEX with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of Figure 3.2-2. In conjunction with the use of the excore monitoring system and in establishing the AXIAL SHAPE INDEX limits, the following assumptions are made: 1) the CEA insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are satisfied, ~~2) the flux peaking augmentation factors are as shown in Figure 4.2-1,~~ 3) 2 the AZIMUTHAL POWER TILT restrictions of Specification 3.2.4 are satisfied, and 4) 3 the TOTAL PLANAR RADIAL PEAKING FACTOR does not exceed the limits of Specification 3.2.2.

The Incore Detector Monitoring System continuously provides a direct measure of the peaking factors and the alarms which have been established for the individual incore detector segments ensure that the peak linear heat rates will be maintained within the allowable limits of Figure 3.2-1. The setpoints for these alarms include allowances, set in the conservative directions, for ~~1) flux peaking augmentation factors as shown in Figure 4.2-1,~~ 2) 1 a measurement-calculational uncertainty factor of 1.070 1.062, 3) 2 an engineering uncertainty factor of 1.03, 4) 3 an allowance of 1.01 1.002 for axial fuel densification and thermal expansion, and 5) 4 a THERMAL POWER measurement uncertainty factor of 1.02.

3/4.2.2, 3/4.2.3 and 3/4.2.4 TOTAL PLANAR AND INTEGRATED RADIAL PEAKING FACTORS - F^T_{xy} AND F^T_r AND AZIMUTHAL POWER TILT - T_q

The limitations on F^T_{xy} and T_q are provided to ensure that the assumptions used in the analysis for establishing the Linear Heat Rate and Local Power Density - High LCOs and LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. The limitations on F^T_r and T_q are provided to ensure that the assumptions used in

BASES

the analysis establishing the DNB Margin LCO, and Thermal Margin/Low Pressure LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. If F^T_{xy}, F^T_r or T_q exceed their basic limitations, operation may continue under the additional restrictions imposed by the ACTION statements since these additional restrictions provide adequate provisions to assure that the assumptions used in establishing the Linear Heat Rate, Thermal Margin/Low Pressure and Local Power Density - High LCOs and LSSS setpoints remain valid. An AZIMUTHAL POWER TILT > 0.10 is not expected and if it should occur, subsequent operation would be restricted to only those operations required to identify the cause of this unexpected tilt.

The value of T_q that must be used in the equation $F^T_{xy} = F_{xy}(1 + T_q)$ and $F^T_r = F_r(1+T_q)$ is the measured tilt.

The surveillance requirements for verifying that F^T_{xy}, F^T_r and T_q are within their limits provide assurance that the actual values of F^T_{xy}, F^T_r and T_q do not exceed the assumed values. Verifying F^T_{xy} and F^T_r after each fuel loading prior to exceeding 75% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the safety analyses assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.23 throughout each analyzed transient.

1.21

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The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

Ⓐ

In addition to the DNB criteria, there are two other criteria which set the specification in Figure 3.2-4. The second criteria is to ensure that the existing core power distribution at full power is less severe than the power distribution factored into the small-break LOCA analysis. This results in a limitation on the allowed negative AXIAL SHAPE INDEX value at full power. The third criteria is to maintain limitations on peak linear heat rate at low power levels resulting from Anticipated Operational Occurrences (AOOs). Figure 3.2-4 is used to assure the LHR criteria for this condition because the linear heat rate LCO, for both ex-core and in-core monitoring, is set to maintain only the LOCA kw/ft requirements which are limiting at high power levels. At reduced power levels, the kw/ft requirements of certain AOOs (e.g., CEA withdrawal), tend to become more limiting than that for LOCA.

10.0 STARTUP TESTING

The startup testing program proposed for Cycle 7 is identical to the program proposed for the reference cycle in Reference 1.

11.0 REFERENCES

References - Chapters 1 Through 3

1. Letter, A. E. Lundvall, Jr. (BG&E) to J. R. Miller (NRC), Docket No. 50-317, "Calvert Cliffs Unit 1 Eighth Cycle License Application," February 22, 1985.

2. Calvert Cliffs Nuclear Power Plant Units 1 and 2 Updated Final Safety Analysis Report Chapter 3.

3. Letter, D. H. Jaffe (NRC) to A. E. Lundvall, Jr. (BG&E), Docket No. 50-317, "Safety Evaluation of Calvert Cliffs Unit 1 Cycle 8," May 20, 1985.

4. Letter, A. E. Lundvall, Jr. (BG&E) to R. A. Clark (NRC), Docket Nos. 50-317 and 50-318, "Topical Report for Extended Burnup Operation of C-E Fuel," June 7, 1982; Enclosure CENPD-269-P, "Extended Burnup Operation of Combustion Engineering PWR Fuel," April 1982.

5. Letter, A. E. Lundvall, Jr. (BG&E) to T. E. Murley (NRC), "Calvert Cliffs Nuclear Power Plant Unit No. 2, Docket No. 50-318 Report of Startup Testing for Cycle 6," October 8, 1984.

6. CEN-105(B)-P, "Reconstitutable B_4C Type CEA Design for Use in the BG&E Reactor," February, 1979.

7. CE-NPSD-225-P, "Results of Calvert Cliffs Unit 2 EOC-4 Poolside Inspection of Control Element Assemblies," April, 1983.

8. CE-NPSD-266-P, "Results of Calvert Cliffs-2 EOC-5 Poolside Inspection of Control Element Assemblies," May, 1985.

References - Chapter 4

1. Letter, A. E. Lundvall, Jr. (BG&E) to J. R. Miller (NRC), Docket No. 50-317, "Calvert Cliffs Unit 1 Eighth Cycle License Application," February 22, 1985.

2. Calvert Cliffs Nuclear Power Plant Units 1 and 2 Updated Final Safety Analysis Report Chapter 3.

3. Letter, A. E. Lundvall, Jr. (BG&E) to R. A. Clark (NRC), Docket No. 50-318, "Calvert Cliffs Unit 2 Fifth Cycle License Application," October 15, 1982.

4. Letter, A. E. Lundvall, Jr. (BG&E) to R. W. Reid (NRC), Docket No. 50-317, "Proposed Finding of No Unreviewed Safety Question on Unit 2, Cycle 3 Reload Core Design," July 11, 1979.

5. Letter, A. E. Lundvall, Jr. (BG&E) to R. W. Reid (NRC), Docket No. 50-318, "Unit 2 Cycle 2 License Application," July 26, 1978.

6. Letter, A. E. Lundvall, Jr. (BG&E) to J. R. Miller (NRC), Docket Nos. 50-317 & 50-318, "Request for Amendment," (Clad Collapse/Augmentation Factors), December 31, 1984.

7. EPRI Report NP-3966-CCM, Volume 5, "Evaluation of Interpellet Gap Formation and Clad Collapse in Modern PWR Fuel Rods," April 1985.

8. Letter, D. H. Jaffe (NRC) to A. E. Lundvall, Jr. (BG&E), Docket No. 50-317, "Safety Evaluation of Calvert Cliffs Unit 1 Cycle 8," May 20, 1985.

9. CEN-183(B)-P, "Application of CENPD-198 to Zircaloy Component Dimensional Changes," September 1981.

10. Letter, D. H. Jaffe (NRC) to A. E. Lundvall, Jr. (BG&E), "Regarding Unit 1 Cycle 6 License Approval (Amendment #71 to DPR-53 and SER)," June 24, 1982.

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