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ATTACHMENT 1

A. DESCRIPTION OF THE TECHNICAL SPECIFICATION AMENDMENT REQUEST

The proposed amendment changes the Shutdown Margin versus Cold Leg Temperature curve as set forth in Technical Specification (T.S.) 3.1.1.2. The change is to the Hot Zero Power endpoint. The change is from 6.0% $\Delta\rho$ to 6.5% $\Delta\rho$.

B. PURPOSE OF THE TECHNICAL SPECIFICATION

The purpose of Technical Specification 3.1.1.2 is to ensure that an adequate shutdown margin is maintained in the reactor at all times.

C. NEED FOR THE TECHNICAL SPECIFICATION AMENDMENT

Due to the design of Cycle 2, the Cycle 2 moderator temperature reactivity insertion is more adverse than Cycle 1 during a postulated steam line break. Because of the more adverse cooldown reactivity insertion for Cycle 2, the Shutdown Margin is required to be increased from 6% $\Delta\rho$ to 6.5% $\Delta\rho$ at zero power. The increase in margin is required to maintain the operation of Cycle 2 within the safety analysis.

D. BASIS FOR PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

1. The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a proposed amendment would not: (1) Involve a significant increase in the probability of consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety.

   A discussion of these standards as they relate to the amendment request follows:

   Standard 1--Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed change ensures that the analysis of the most limiting accident, the Steam Line Break event for Cycle 2, is bounded by the reference cycle (Cycle 1) transient analysis. Therefore, there is no increase in the probability or consequences of an accident previously evaluated because operation of Cycle 2 is within the realm of operation, as experienced during Cycle 1.

Standard 2--Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated because, by increasing the required shutdown margin at zero power, the Cycle 2 transient analysis is bounded by the reference cycle transient analysis. Requiring a larger shutdown margin does not subject the operation of Cycle 2 to any additional accidents. It restricts the Unit even further in its allowed operation. Therefore, there will be no increase in the possibility of a new or different kind of accident occurring.

Standard 3--Involve a significant reduction in a margin of safety.

The proposed change does not involve a significant reduction in a margin of safety because the shutdown margin at zero power is being increased to ensure the same margin of safety is maintained for Cycle 2 operation as it was for Cycle 1. The increased shutdown margin ensures that the most limiting event is bounded by the reference cycle transient analysis and thus maintaining margin.

2. The proposed amendment matches the guidance concerning the application of standards for determining whether or not a significant hazards consideration exists (51 FR 7751) by example:

(iii) For a nuclear power reactor, a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptable criteria for the Technical Specifications, the analytical methods used to demonstrate conformance with the Technical Specifications and regulations are not significantly changed, and that NRC has previously found such methods acceptable.

E. SAFETY EVALUATION FOR THE AMENDMENT REQUEST

The proposed Technical Specification amendment will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR. The proposed change does not change or replace equipment or components important to safety. The change ensures that, during the operation of Cycle 2, the Cycle 2 analysis is bounded by the reference cycle transient analysis. Therefore, there is no increase in the probability of occurrence of the consequences of an accident or malfunction of equipment.

The proposed Technical Specification amendment will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR. The proposed change ensures that, during the operation of Cycle 2, a shutdown margin of the same magnitude as the margin required

during Cycle 1 is maintained. By increasing the margin to 6.5% ,the Cycle 2 analysis is bounded by the reference cycle transient analysis and restricts the Unit even further in its allowed operation. Therefore, there is no increase in the possibility for an accident or malfunction being created.

The proposed Technical Specification amendment will not reduce the margin of safety as defined in the basis for the Technical Specifications. The proposed change ensures that during the operation of Cycle 2, the Cycle 2 analysis is bounded by the reference cycle transient analysis and, therefore, there is no reduction in the margin.

F. ENVIRONMENTAL IMPACT CONSIDERATION DETERMINATION

The proposed change request does not involve an unreviewed environmental question because operation of PVNGS Unit 2, in accordance with this change, would not:

1. Result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the staff's testimony to the Atomic Safety and Licensing Board; or

2. Result in a significant change in effluents or power levels; or

3. Result in matters not previously reviewed in the licensing basis for PVNGS which may have a significant environmental impact.

G. MARKED-UP TECHNICAL SPECIFICATION CHANGE PAGES

Limiting Conditions For Operation And Surveillance Requirements:

3/4 1-2a

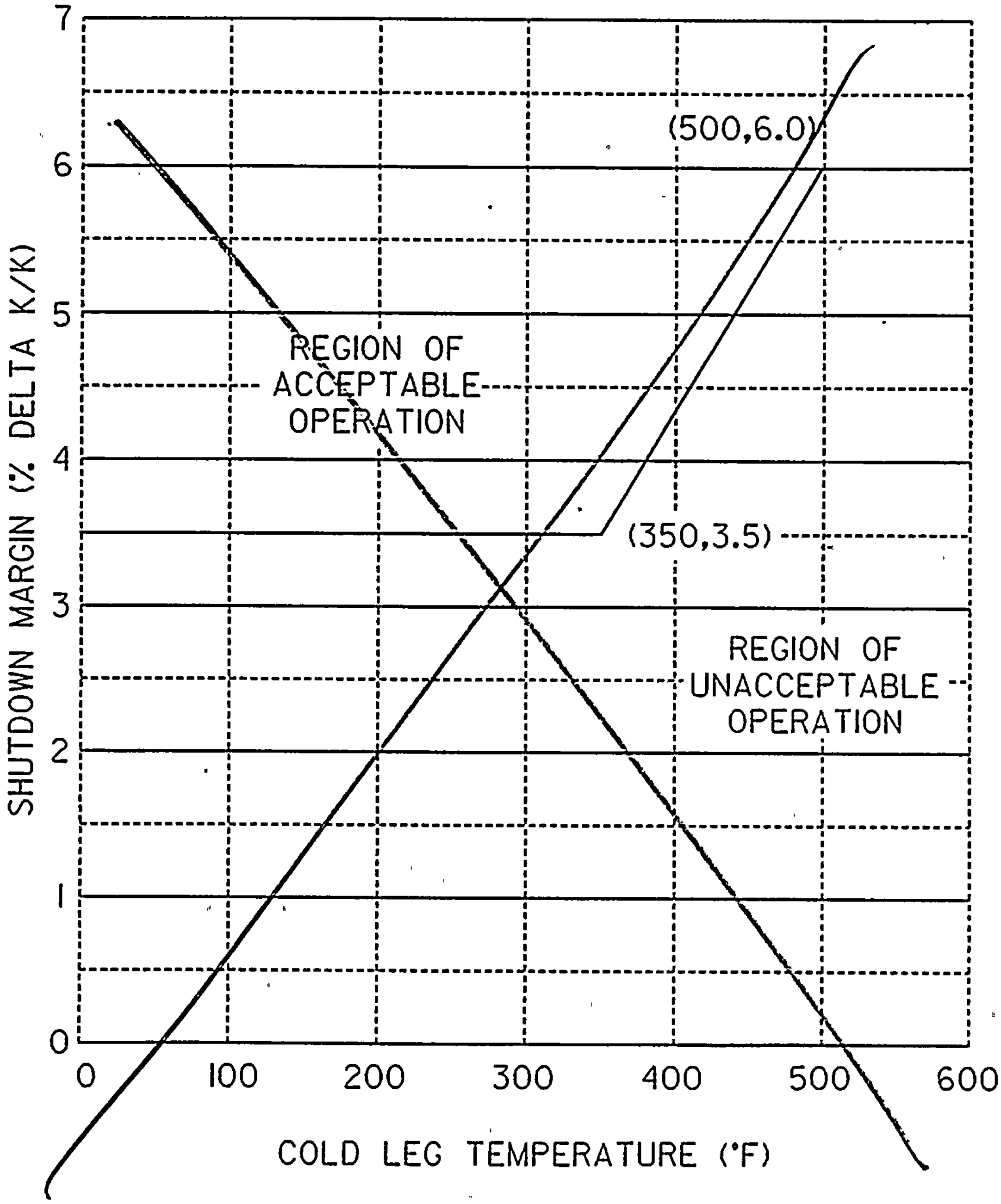


FIGURE 3.1-1A

SHUTDOWN MARGIN VERSUS COLD LEG TEMPERATURE

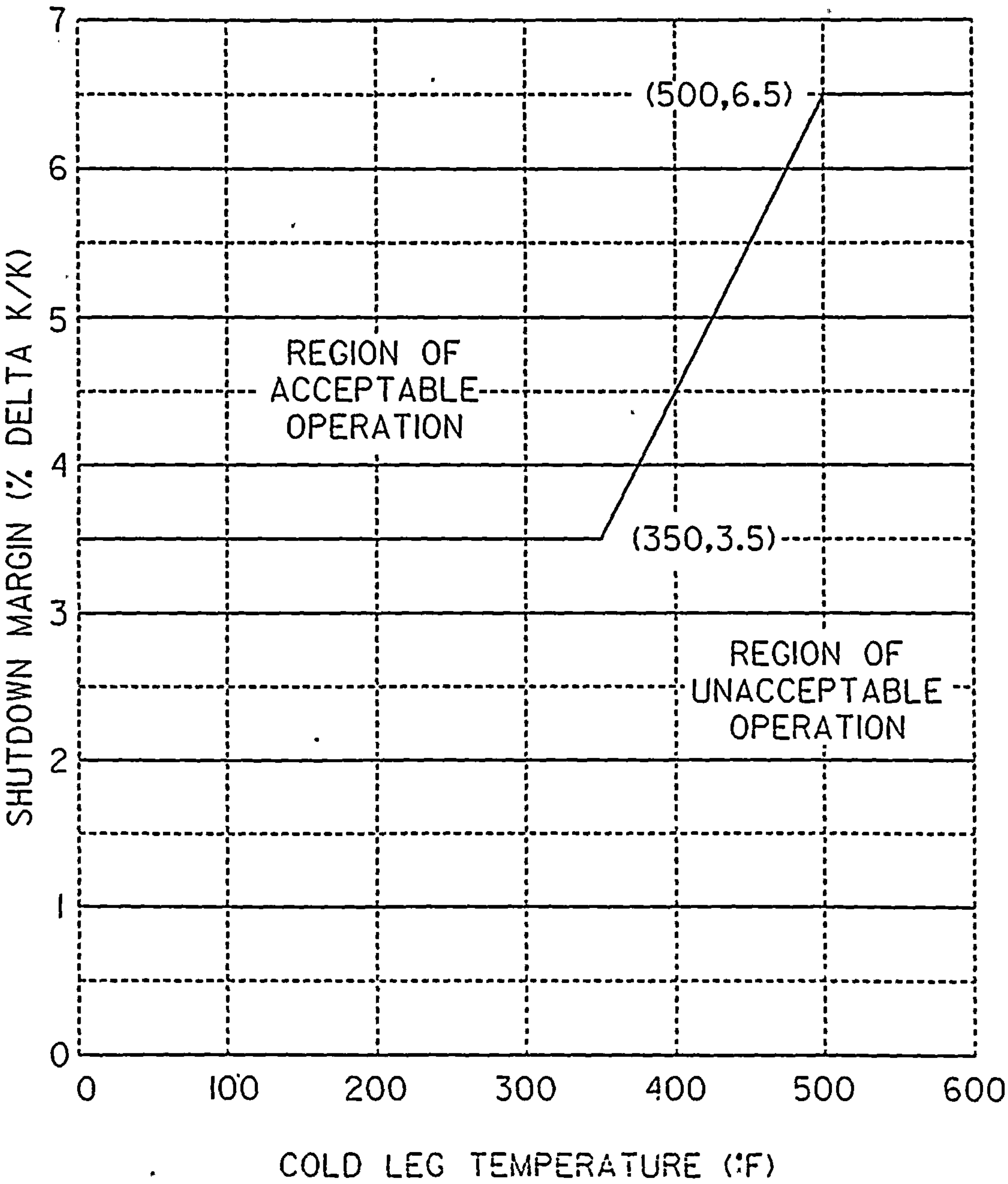


FIGURE 3.1-1A

SHUTDOWN MARGIN VERSUS COLD LEG TEMPERATURE

ATTACHMENT 2

A. DESCRIPTION OF THE TECHNICAL SPECIFICATION AMENDMENT REQUEST

The proposed amendment changes the Moderator Temperature Coefficient (MTC) Figure 3.3-1 as set forth in Technical Specification (T.S.) 3.1.1.3. The changes are two fold. The operating bounds of the MTC are being broadened to accommodate the operation of Cycle 2 and the x axis is being changed to core power level instead of average moderator temperature.

B. PURPOSE OF THE TECHNICAL SPECIFICATION

T.S. 3.1.1.3 ensures that the assumptions used in the accident and transient analysis remain valid through each fuel cycle.

C. NEED FOR THE TECHNICAL SPECIFICATION AMENDMENT

In preparation for future 18 months cycles, the Cycle 2 core physics is such that a change in the MTC operating band will occur. To accommodate operation throughout Cycle 2, the MTC operating band has become more positive because of the increase in fuel enrichment which requires higher boron concentration at beginning of the cycle. As operation into the cycle proceeds, the MTC will become more negative. In addition, the x axis is to be changed to core power level instead of average moderator temperature. By changing the x axis to core power level, the method of calculating the bounding MTC for the most limiting case becomes simplified. Making the MTC a dependent variable of core power only and not of inlet temperature and core power, as the present curve represents, the calculation of the limiting MTC need only be performed once. The present method of manipulating MTC requires performing the analyses several times at various average moderator temperatures to be sure of obtaining the most limiting case but, with the new method, MTC can be calculated once and there is assurance that the most limiting case value is obtained. Both graphs are the results of the same set of codes, only the method of manipulating the data is slightly different.

D. BASIS FOR PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

1. The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92 A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety.

A discussion of these standards as they relate to the amendment request follows:

Standard 1--Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated because the consequences of any accident, when the unit is operated in the calculated band of the Cycle 2 MTC, is bounded by the reference Cycle (Cycle 1) transient analysis. Therefore, there is no possibility of an accident previously evaluated being increased.

Standard 2--Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated. The results of the analysis performed for Cycle 2, using the proposed MTC band, assures that there will be sufficient margin for the most limiting DBE. By operating within these limits, operation of Cycle 2 will not create any situation where a new or different kind of accident could occur because Cycle 2 analysis results show that Cycle 2 is bounded by the reference cycle analysis.

Standard 3--Involve a significant reduction in a margin of safety.

The proposed change does not involve a significant reduction in a margin of safety because the results for all DBEs affected by the new MTC are bounded by the reference analysis. Therefore, the margin of safety does not change.

2. The proposed amendment matches the guidance concerning the application of standards for determining whether or not a significant hazards consideration exists (51 FR 7751) by example:

   (iii) For a nuclear power reactor, a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptable criteria for the technical specifications, the analytical methods used to demonstrate conformance with the technical specifications and regulations are not significantly changed, and that NRC has previously found such methods acceptable.

E. SAFETY EVALUATION FOR THE AMENDMENT REQUEST

The proposed Technical Specification amendment will not increase the probability of occurrence of the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR. The proposed change does not change or replace equipment or components important to safety. The proposed change is still bounded by the reference cycle transient analysis and, therefore, the probability of any accident previously evaluated has not changed.

The proposed Technical Specification amendment will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR. The results of the analysis performed for Cycle 2, using the MTC band as stated in Fig 3.3-1, assure that there is sufficient margin for the most limiting Design Basis Event (DBE). By operating within these limits, operation of Cycle 2 will not create any situation where a new or different kind of accident could occur because Cycle 2 analysis results show that Cycle 2 is bounded by the reference cycle analysis.

The proposed Technical Specification amendment will not reduce the margin of safety as defined in the basis for the technical specifications. The results for all DBEs affected by the new MTC are bounded by the reference analysis.

F. <u>ENVIRONMENTAL IMPACT CONSIDERATION DETERMINATION</u>

The proposed change request does not involve an unreviewed environmental question because operation of PVNGS Unit 2, in accordance with this change, would not:

1. Result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the staff's testimony to the Atomic Safety and Licensing Board; or

2. Result in a significant change in effluents or power levels; or

3. Result in matters not previously reviewed in the licensing basis for PVNGS which may have a significant environmental impact.

G. <u>MARKED-UP TECHNICAL SPECIFICATION CHANGE PAGES</u>

Limiting Condition for Operation and Surveillance Requirements:

3/4 1-5

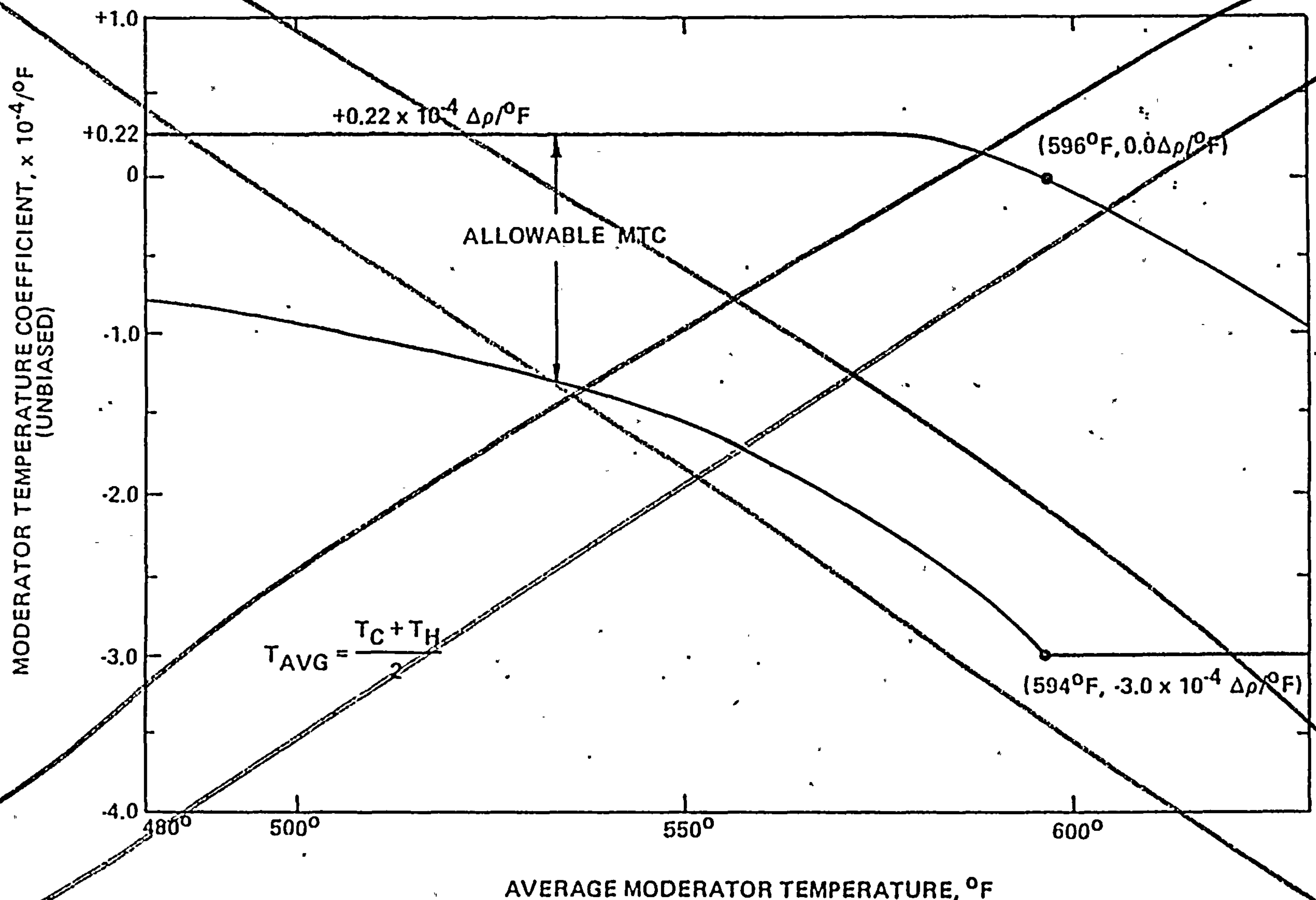
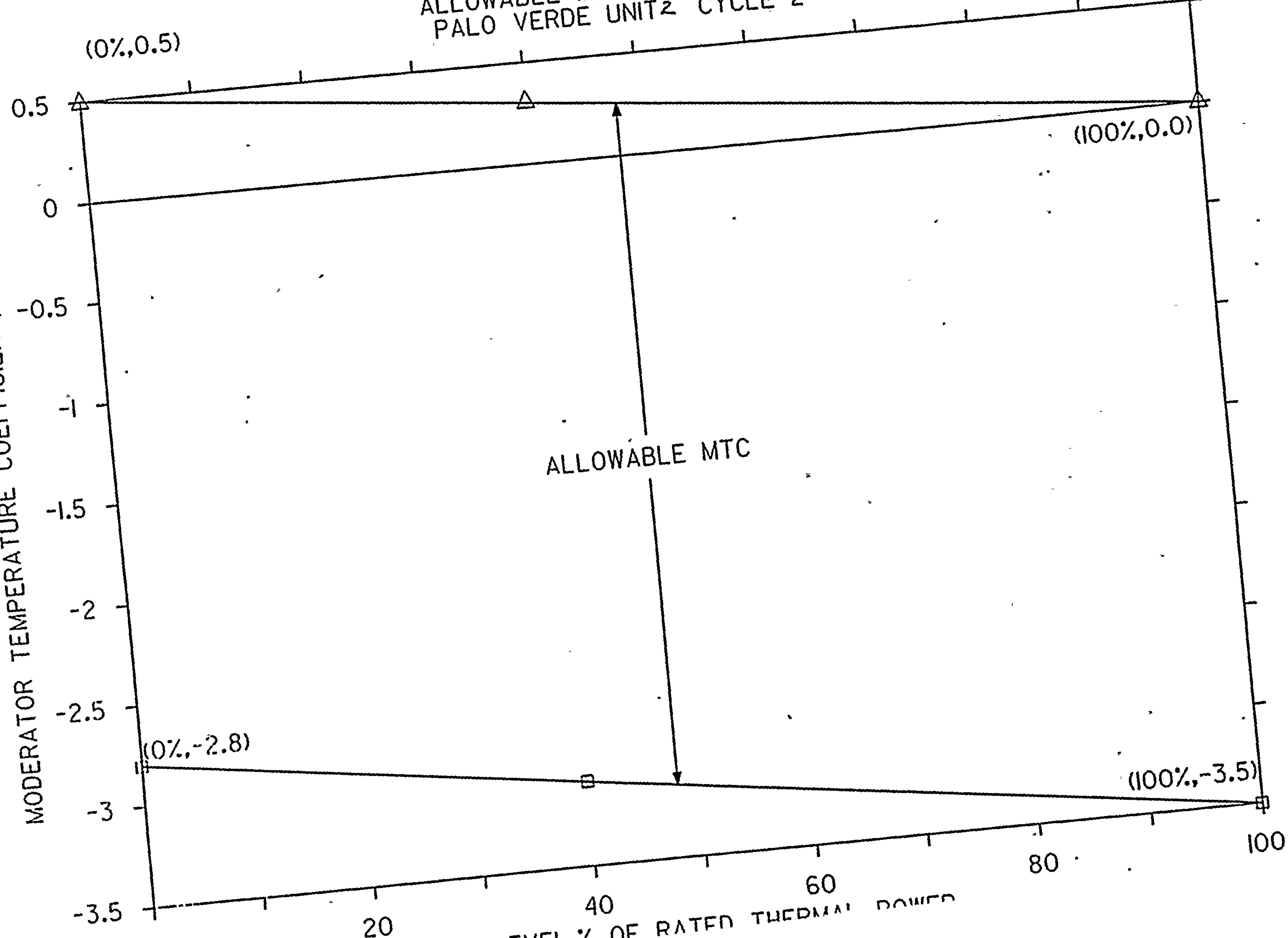


FIGURE 3.1-1

ALLOWABLE MTC MODES 1 AND 2 PALO VERDE UNIT 2 CYCLE 1

FIGURE 3.1-1
ALLOWABLE MTC MODES 1 AND 2
PALO VERDE UNIT 2 CYCLE 2



ATTACHMENT 3

A. DESCRIPTION OF THE TECHNICAL SPECIFICATION AMENDMENT REQUEST

The proposed amendment changes the operational pressure band of the pressurizer, as set forth in Technical Specification (T.S.) 3.2.8 to a tighter operational band. The band is being changed from 1815 psia thru 2370 psia to 2025 psia thru 2300 psia.

B. PURPOSE OF THE TECHNICAL SPECIFICATION

T.S. 3.2.8 ensures that the actual value of pressurizer pressure is maintained within the range of values used in the safety analyses.

C. NEED FOR THE TECHNICAL SPECIFICATION AMENDMENT

To support the Core Protection Calculator (CPC) Improvement Program, the operational pressure band of the pressurizer requires tightening. Potential transients initiated at the extremes of the Cycle 1 pressure range were not analyzed for Cycle 2. Because the calculations were not performed, the CPCs cannot support normal operation outside of the proposed pressurizer pressure band.

D. BASIS FOR PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

1. The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) Involve a significant reduction in a margin of safety.

   A discussion of these standards as they relate to the amendment request follows:

   Standard 1--Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated because the change ensures maintaining the safety margin, as required by the reference cycle (Cycle 1) safety analysis or the safety limits as stated in the FSAR. The change restricts normal operation because there are no supporting calculations and related penalty factors for normal operation outside the specified pressure range. The bounds of the safety analysis have not been changed. Therefore, there will be no increase in the possibility or consequences of an accident.

Standard 2--Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated because the change ensures that the safety margin as required by the reference cycle safety analysis is maintained. Since the operation band is more restrictive in relation to the safety analysis it can be concluded that there will be no increase in the possibility of a new or different kind of accident.

Standard 3--Involve a significant reduction in a margin of safety.

The proposed change does not involve a significant reduction in a margin of safety because the proposed change ensures maintaining the safety margin as required by the reference cycle safety analysis or the safety limits as stated in the FSAR. By reducing the operation band of the pressurizer, initial conditions during an accident are more restricted but, because the bounds of the safety analysis have not changed, the margin of safety has not been reduced.

2. The proposed amendment matches the guidance concerning the application of standards for determining whether or not a significant hazards consideration exists (51 FR 7751) by example:

(iii) For a nuclear power reactor, a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptable criteria for the Technical Specifications, the analytical methods used to demonstrate conformance with the Technical Specifications and regulations are not significantly changed, and that NRC has previously found such methods acceptable.

E. SAFETY EVALUATION FOR THE AMENDMENT REQUEST

The proposed Technical Specification amendment will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR. The proposed change does not change or replace equipment or components important to safety. The proposed change ensures that the safety margin as required by the reference cycle safety analysis is maintained. The change restricts normal operation because there are no supporting calculations and related penalty factors for normal operation outside the specified pressure range. The bounds of the safety analysis have not been changed.

The proposed Technical Specification amendment will not create the possibility for an accident or malfunction of a different type than any previously

evaluated in the FSAR. The proposed change ensures that the safety margin as required by the reference cycle safety analysis is maintained. Since the operation band is more restrictive in relation to the safety analysis, it can be concluded that there will be no increase in the possibility of a new or different kind of accident.

The proposed Technical Specification amendment will not reduce the margin of safety as defined in the basis for the Technical Specifications. The proposed change ensures that the safety margin as required by the reference cycle safety analysis is maintained. By reducing the operation band of the pressurizer, initial conditions during an accident are more restricted but, because the bounds of the safety analysis have not changed, the margin of safety has not been reduced.

F. ENVIRONMENTAL IMPACT CONSIDERATION DETERMINATION

The proposed change request does not involve an unreviewed environmental question, because operation of PVNGS Unit 2, in accordance with this change would not:

1. Result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the staff's testimony to the Atomic Safety and Licensing Board; or

2. Result in a significant change in effluents or power levels; or

3. Result in matters not previously reviewed in the licensing basis for PVNGS which may have a significant environmental impact.

G. MARKED-UP TECHNICAL SPECIFICATION CHANGE PAGES

Limiting Conditions For Operation And Surveillance Requirements:

3/4 2-12

3/4.2.8 PRESSURIZER PRESSURE

LIMITING CONDITION FOR OPERATION

2025

3.2.8 The pressurizer pressure shall be maintained between ~~1815~~ psia and
2300 ~~2370~~ psia.

APPLICABILITY: MODES 1 and 2*.

ACTION:

With the pressurizer pressure outside its above limits, restore the pressure to within its limit within 2 hours or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.8 The pressurizer pressure shall be determined to be within its limit at least once per 12 hours.

*See Special Test Exception 3.10.5

ATTACHMENT 4

A. DESCRIPTION OF THE TECHNICAL SPECIFICATION AMENDMENT REQUEST

The proposed amendment modifies the CEA position Technical Specifications (T.S.) 3.1.3.1 and 3.1.3.2 by removing direct references of the control of insertion of the Part-length Control Element Assemblies (PLCEA) and creates an additional T.S. that addresses the length of time for insertion and the insertion limit of the PLCEA specifically.

B. PURPOSE OF THE TECHNICAL SPECIFICATION

The purpose of T.S. 3.1.3.1 and 3.1.3.2 is to ensure that (1) acceptable power distribution limits are maintained, (2) the minimum shutdown margin is maintained, and (3) the potential effects of CEA misalignments are limited to acceptable levels.

C. NEED FOR TECHNICAL SPECIFICATION AMENDMENT

Creating a separate T.S. for addressing operation of the PLCEA would provide an improvement to the potential consequences of a PLCEA drop or slip initiated from an allowable inserted position. It would also add a more explicit Limiting Condition for Operation to clarify the allowable duration for the PLCEA to remain within the defined ranges of axial position.

D. BASIS FOR PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

1. The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident previously evaluated; or (3) Involve a significant reduction in a margin of safety.

   A discussion of these standards as they relate to the amendment request follows:

   Standard 1--Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed change provides additional assurance that adverse axial shapes and rapid local power changes, which affect radial power peaking

factors and DNB considerations, do not occur as a result of the part length CEA group being positioned in the same axial segment of fuel assemblies for an extended period of time during operation. Because the proposed change will impose more restrictive limits along with surveillance requirements to ensure adherence with the insertion limits, this proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

Standard 2--Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed change provides additional assurance that adverse axial shapes and rapid local power changes, which affect radial power peaking factors and DNB considerations, do not occur as result of the part-length CEA group being positioned in the same axial segment of fuel assemblies for an extended period of time during operation. Because the proposed change will impose more restrictive limits with respect to previously analyzed events, along with surveillance requirements to ensure adherence with the insertion limits, this proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Standard 3--Involve a significant reduction in a margin of safety.

The proposed change does not involve a significant reduction in a margin of safety because the proposed change provides additional assurance that adverse axial shapes and rapid local power changes, which affect radial power peaking factors and DNB considerations, do not occur as a result of the part-length CEA group being positioned in the same axial segment of fuel assemblies for an extended period of time during operation. Because the proposed change will impose more restrictive limits along with surveillance requirements to ensure adherence with the insertion limits, this proposed change does not involve a significant reduction in the margin of safety.

2. The proposed amendment matches the guidance concerning the application of standards for determining whether or not a significant hazards consideration exists (51 FR 7751) by example:

   (ii) A change constitutes an additional limitation, restriction or control not presently included in the Technical Specifications: for example, a more stringent surveillance requirement.

E. SAFETY EVALUATION FOR THE AMENDMENT REQUEST

The proposed Technical Specification amendment will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR. The proposed change does not change or replace equipment or components important to safety.

The proposed change provides additional assurance that adverse axial shapes and rapid local power changes, which affect radial power peaking factors and DNB considerations, do not occur as a result of the part-length CEA group being positioned in the same axial segment of fuel assemblies for an extended period of time during operation. Because the proposed change will impose more restrictive limits, along with surveillance requirements to ensure adherence with the insertion limits, this proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

The proposed Technical Specification amendment will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR. The proposed change provides additional assurance that adverse axial shapes and rapid local power changes, which affect radial power peaking factors and DNB considerations, do not occur as a result of the part-length CEA group being positioned in the same axial segment of fuel assemblies for an extended period of time during operation. Because the proposed change will impose more restrictive limits along with surveillance requirements to ensure adherence with the insertion limits, this proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

The proposed Technical Specification amendment will not reduce the margin of safety as defined in the basis for the technical specifications. The proposed change provides additional assurance that adverse axial shapes and rapid local power changes, which affect radial power peaking factors and DNB considerations, do not occur as a result of the part-length CEA group being positioned in the same axial segment of fuel assemblies for an extended period of time during operation. Because the proposed change will impose more restrictive limits, along with surveillance requirements to ensure adherence with the insertion limits, this proposed change does not involve a significant reduction in a margin of safety.

F. ENVIRONMENTAL IMPACT CONSIDERATION DETERMINATION

The proposed change request does not involve an unreviewed environmental question because operation of PVNGS Unit 2, in accordance with this change, would not:

1. Result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the staff's testimony to the Atomic Safety and Licensing Board; or

2. Result in a significant change in effluents or power levels; or

3. Result in matters not previously reviewed in the licensing basis for PVNGS which may have a significant environmental impact.

G. <u>MARKED-UP TECHNICAL SPECIFICATION CHANGE PAGES</u>

Limiting Conditions For Operation and Surveillance Requirements:

	3/4 1-21	XIX
	3/4 1-22	IV
	3/4 1-23	3/4 1-1
	3/4 1-24	3/4 1-2
	3/4 1-25	3/4 10-2
B	3/4 1-6	3/4 10-4
B	3/4 1-7	

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIST OF FIGURES

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

CEA POSITION

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full-length (shutdown and regulating) CEAs, and all part-length CEAs which are inserted in the core, shall be OPERABLE with each CEA of a given group positioned within 6.6 inches (indicated position) of all other CEAs in its group. ~~In addition, the position of the part length CEAs Groups shall be limited to the insertion limits shown in Figure 3.1-2A~~.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more full-length CEAs inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.~~1~~ 2 is satisfied within 1 hour and be in at least HOT STANDBY within 6 hours.

b. With more than one full-length or part-length CEA inoperable or misaligned from any other CEA in its group by more than 19 inches (indicated position), be in at least HOT STANDBY within 6 hours.

c. With one or more full-length or part-length CEAs misaligned from any other CEAs in its group by more than 6.6 inches, operation in MODES 1 and 2 may continue, provided that core power is reduced in accordance with Figure 3.1-2~~B~~ A and that within 1 hour the misaligned CEA(s) is either:

1. Restored to OPERABLE status within its above specified alignment requirements, or

2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.~~1~~ 2 is satisfied. After declaring the CEA(s) inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specifications 3.1.3.6 and 3.1.3.7 provided:

a) Within 1 hour the remainder of the CEAs in the group with the inoperable CEA(s) shall be aligned to within 6.6 inches of the inoperable CEA(s) while maintaining the allowable CEA sequence and insertion limits shown on Figures 3.1-2A, 3.1-3 and 3.1-4; the THERMAL POWER level shall be restricted pursuant to Specifications 3.1.3.6 and 3.1.3.7 during subsequent operation.

*See Special Test Exceptions 3.10.2 and 3.10.4.

CONTROLLED BY USER

ACTION: (Continued)

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.~~1~~ 2 is determined at least once per 12 hours.

Otherwise, be in at least HOT STANDBY within 6 hours.

d. With one full-length CEA inoperable due to causes other than addressed by ACTION a., above, but within its above specified alignment requirements, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6.

e. With one part-length CEA inoperable and inserted in the core, operation may continue provided the alignment of the inoperable part length CEA is maintained within 6.6 inches (indicated position) of all other part-length CEAs in its group~~.~~ and the CEA is maintained pursuant to the requirements of Specification 3.1.3.7

Deleted

~~f. With part length CEAs inserted beyond insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 2 hours either:~~

~~1. Restore the part length CEAs to within their limits, or~~

~~2. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by part length CEA group position using Figure 3.1-2A.~~

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full-length and part-length CEA shall be determined to be within 6.6 inches (indicated position) of all other CEAs in its group at least once per 12 hours except during time intervals when one CEAC is inoperable or when both CEACs are inoperable, then verify the individual CEA positions at least once per 4 hours.

4.1.3.1.2 Each full-length CEA not fully inserted and each part-length CEA which is inserted in the core shall be determined to be OPERABLE by movement of at least 5 inches in any one direction at least once per 31 days.

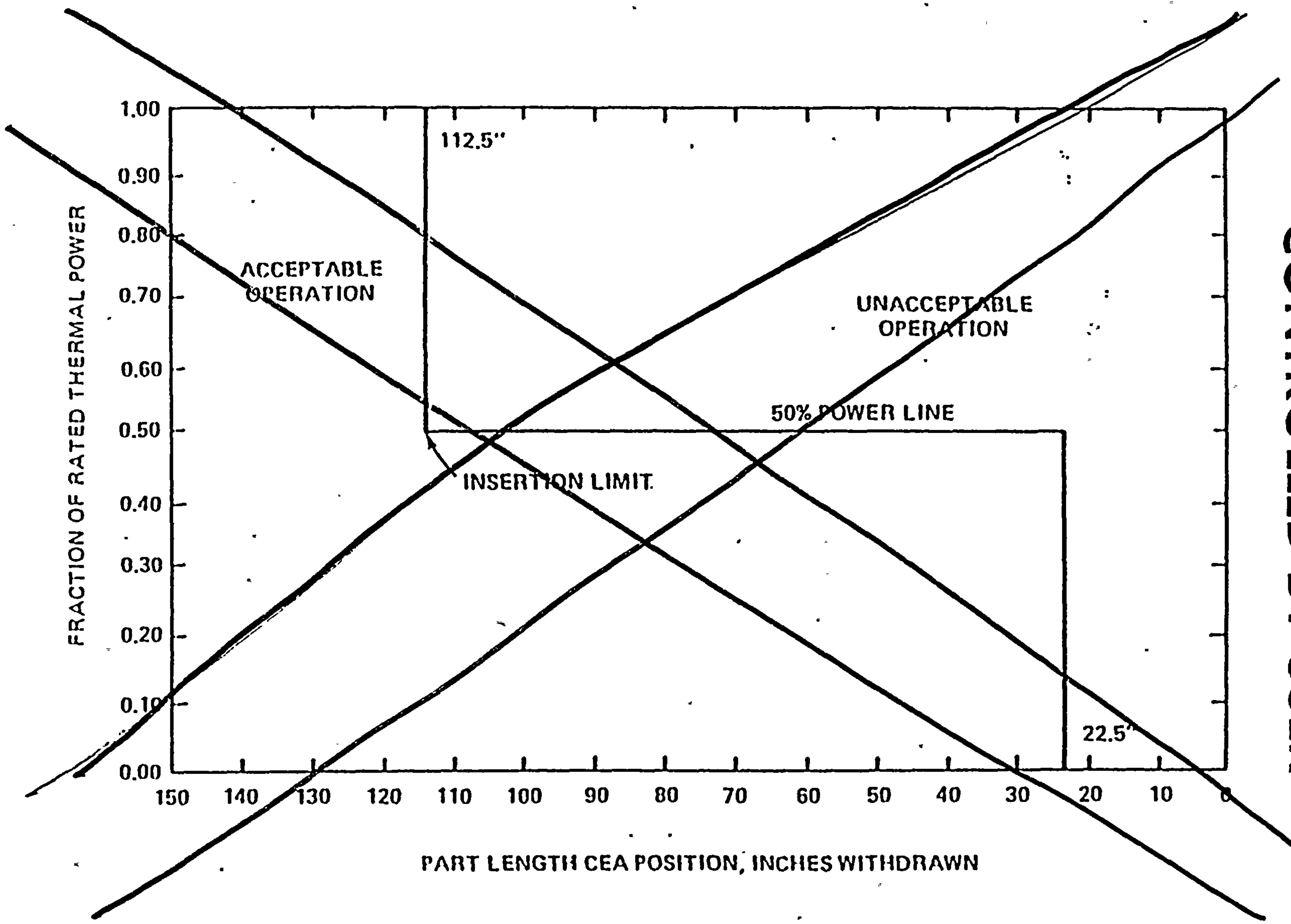

FIGURE 3.1-2A

PART LENGTH CEA INSERTION LIMIT VS. THERMAL POWER

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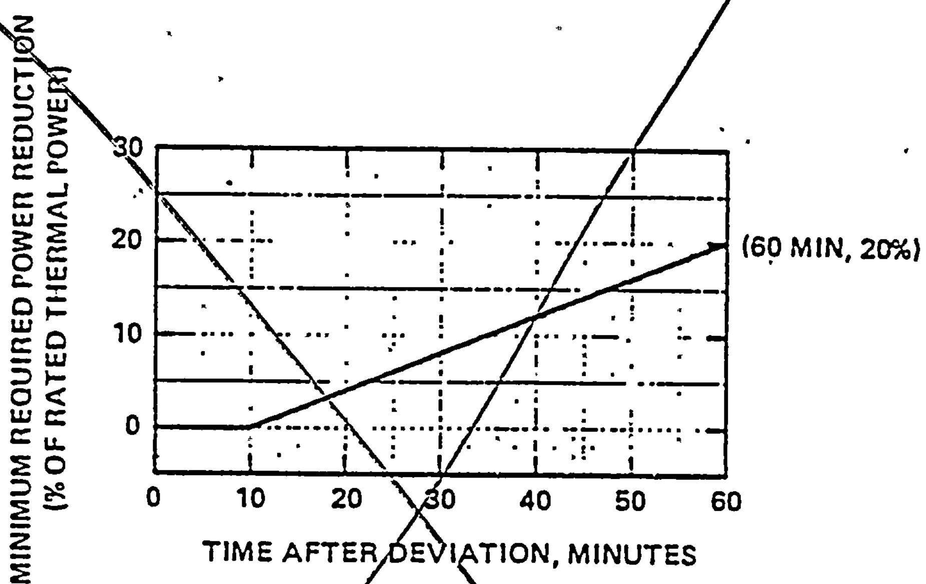
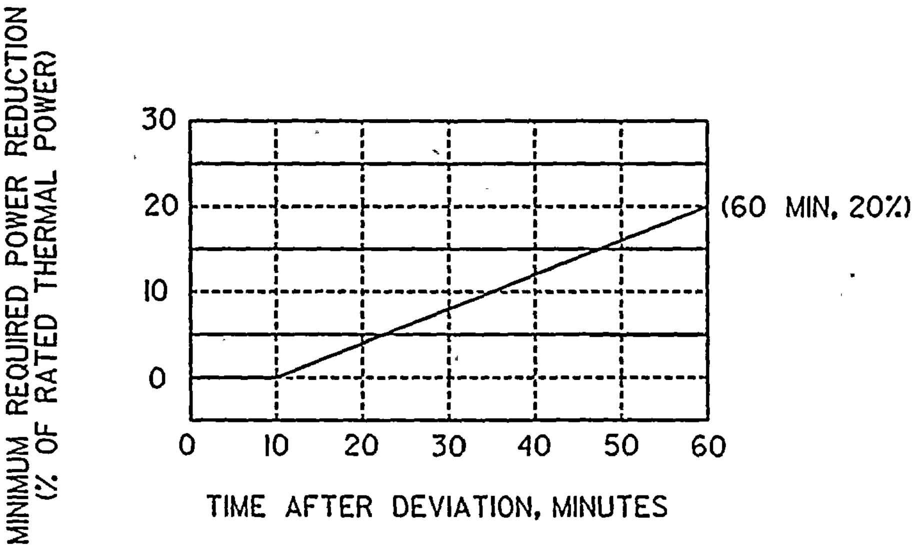


*WHEN CORE POWER IS REDUCED TO 55% OF RATED THERMAL POWER PER THIS LIMIT CURVE, FURTHER REDUCTION IS NOT REQUIRED

FIGURE 3.1-2~~X~~ A

CORE POWER LIMIT AFTER CEA DEVIATION*

FIGURE 3.1.2A
CORE POWER LIMIT AFTER CEA DEVIATION*



*WHEN CORE POWER IS REDUCED TO 55% OF RATED THERMAL POWER PER THIS LIMIT CURVE, FURTHER REDUCTION IS NOT REQUIRED

FIGURE 3.1-2A

CORE POWER LIMIT AFTER CEA DEVIATION*

LIMITING CONDITION FOR OPERATION

3.1.3.2 At least two of the following three CEA position indicator channels shall be OPERABLE for each CEA:

a. CEA Reed Switch Position Transmitter (RSPT 1) with the capability of determining the absolute CEA positions within 5.2 inches,

b. CEA Reed Switch Position Transmitter (RSPT 2) with the capability of determining the absolute CEA positions within 5.2 inches, and

c. The CEA pulse counting position indicator channel.

APPLICABILITY: MODES 1 and 2.

ACTION:

With a maximum of one CEA per CEA group having only one of the above required CEA position indicator channels OPERABLE, within 6 hours either:

a. Restore the inoperable position indicator channel to OPERABLE status, or

b. Be in at least HOT STANDBY, or

c. Position the CEA group(s) with the inoperable position indicator(s) at its fully withdrawn position while maintaining the requirements of Specifications 3.1.3.1, ~~and~~ 3.1.3.6 and 3.1.3.7. Operation may then continue provided the CEA group(s) with the inoperable position indicator(s) is maintained fully withdrawn, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2, and each CEA in the group(s) is verified fully withdrawn at least once per 12 hours thereafter by its "Full Out" limit*.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each of the above required position indicator channels shall be determined to be OPERABLE by verifying that for the same CEA, the position indicator channels agree within 5.2 inches of each other at least once per 12 hours.

*CEAs are fully withdrawn (Full Out) when withdrawn to at least 144.75 inches.

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NEW

PART LENGTH CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.7 The part length CEA groups shall be limited to the insertion limits shown on Figure ~~3.1-5~~ 3.1-5 with PLCEA insertion between the Long Term Steady State Insertion Limit and the Transient Insertion Limit restricted to:

a. ≤ 7 EFPD per 30 EFPD interval, and

b. ≤ 14 EFPD per calender year.

APPLICABILITY: MODE 1 above 20% THERMAL POWER. *

ACTION:

a. With the part length CEA groups inserted beyond the Transient Insertion Limit, except for surveillance testing pursuant to Specification 4.1.3.1.2, within two hours, either:

1. Restore the part length CEA group to within the limits, or

2. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the PLCEA group position using Figure ~~3.1-5.~~ 3.1-5.

b. With the part length CEA groups inserted between the Long Term Steady State Insertion Limit and the Transient Insertion Limit for intervals > 7 EFPD per 30 EFPD interval or > 14 EFPD per calendar year, either:

1. Restore the part length group within the Long Term Steady State Insertion Limits within two hours, or

2. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.7 The position of the part length CEA group shall be determined to be within the Transient Insertion Limit at least once per 12 hours.

*See Special Test Exceptions 3.10.2 and 3.10.4.

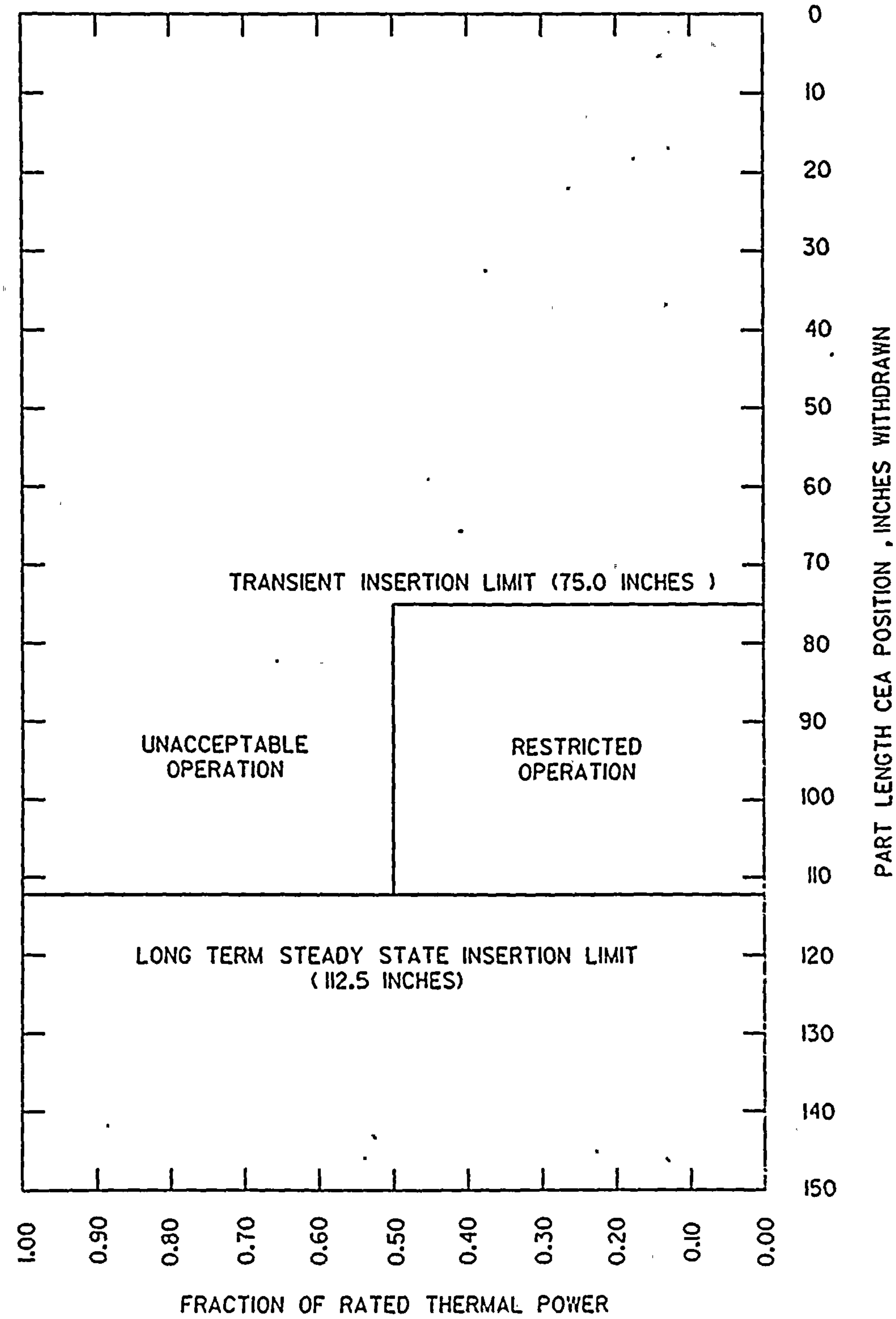

FIGURE 3.1-5
PART LENGTH CEA INSERTION LIMIT VS THERMAL POWER

3/4.10.2 MODERATOR TEMPERATURE COEFFICIENT, GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3,1,3,7,

3.10.2 The moderator temperature coefficient, group height, insertion, and power distribution limits of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, and the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

3,1,3,7,

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, and the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

3,1,3,7,

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, or the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.2 and 3.3.3.2 during PHYSICS TESTS above 20% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, or the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended.

3,1,3,7,

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3/4.10.4 CEA POSITION, REGULATING CEA INSERTION LIMITS AND REACTOR COOLANT COLD LEG TEMPERATURE

,3.1.3.7,

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.1.3.1, 3.1.3.6 and 3.2.6 may be suspended during the performance of PHYSICS TESTS to determine the isothermal temperature coefficient, moderator temperature coefficient, and power coefficient provided the limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.4.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

,3,1,3,7,

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.6 and 3.2.6 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.3.1, 3.1.3.6, 3,1,3,7 and/or 3.2.6 are suspended and shall be verified to be within the test power plateau.

4.10.4.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specification 3.3.3.2 during PHYSICS TESTS above 20% of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1, 3.1.3.6 and/or 3.2.6 are suspended.

,3,1,3,7,

MOVABLE CONTROL ASSEMBLIES (Continued)

and load maneuvering. Analyses are performed based on the expected mode of operation of the NSSS (base load maneuvering, etc.) and from these analyses CEA insertions are determined and a consistent set of radial peaking factors defined. The Long Term Steady State and Short Term Insertion Limits are determined based upon the assumed mode of operation used in the analyses and provide a means of preserving the assumptions on CEA insertions used. The limits specified serve to limit the behavior of the radial peaking factors within the bounds determined from analysis. The actions specified serve to limit the extent of radial xenon redistribution effects to those accommodated in the analyses. The Long and Short Term Insertion Limits of Specifications 3.1.3.6 are specified for the plant which has been designed for primarily base loaded operation but which has the ability to accommodate a limited amount of load maneuvering.

and 3.1.3.7

The Transient Insertion Limits of Specifications 3.1.3.6 and the Shutdown CEA Insertion Limits of Specification 3.1.3.5 ensure that (1) the minimum SHUTDOWN MARGIN is maintained, and (2) the potential effects of a CEA ejection accident are limited to acceptable levels. Long-term operation at the Transient Insertion Limits is not permitted since such operation could have effects on the core power distribution which could invalidate assumptions used to determine the behavior of the radial peaking factors.

The PVNGS CPC and COLSS systems are responsible for the safety and monitoring functions, respectively, of the reactor core. COLSS monitors the DNB Power Operating Limit (POL) and various operating parameters to help the operator maintain plant operation within the limiting conditions for operation (LCO). Operating within the LCO guarantees that in the event of an Anticipated Operational Occurrence (AOO), the CPCs will provide a reactor trip in time to prevent unacceptable fuel damage.

The COLSS reserves the Required Overpower Margin (ROPM) to account for the Loss of Flow (LOF) transient which is the limiting AOO for the PVNGS plants. When the COLSS is Out of Service (COOS), the monitoring function is performed via the CPC calculation of DNBR in conjunction with a Technical Specification COOS Limit Line (Figure 3.2-2) which restricts the reactor power sufficiently to preserve the ROPM.

The reduction of the CEA deviation penalties in accordance with the CEAC (Control Element Assembly Calculator) sensitivity reduction program has been performed. This task involved setting many of the inward single CEA deviation penalty factors to 1.0. An inward CEA deviation event in effect would not be accompanied by the application of the CEA deviation penalty in either the CPC DNB and LHR (Linear Heat Rate) calculations for those CEAs with the reduced penalty factors. The protection for an inward CEA deviation event is thus accounted for separately.

MOVABLE CONTROL ASSEMBLIES (Continued)

If an inward CEA deviation event occurs, the current CPC algorithm applies two penalty factors to each of the DNB and LHR calculations. The first, a static penalty factor, is applied upon detection of the event. The second, a xenon redistribution penalty, is applied linearly as a function of time after the CEA drop. The expected margin degradation for the inward CEA deviation event for which the penalty factor has been reduced is accounted for in two ways. The ROPM reserved in COLSS is used to account for some of the margin degradation. ~~If the combination of the static and xenon redistribution penalties exceeds the reserved ROPM~~ Further, a power reduction in accordance with the curve in Figure 3.1-~~2B~~ 2A is required. In addition, the part length CEA maneuvering is restricted in accordance with Figure 3.1-~~2A~~ 5 to justify reduction of the PLR deviation penalty factors.

The technical specification permits plant operation if both CEACs are considered inoperable for safety purposes after this period.

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ATTACHMENT 5

A. DESCRIPTION OF THE TECHNICAL SPECIFICATION AMENDMENT REQUEST

The proposed amendment changes the response time of the DNBR -Low Reactor Coolant Pump (RCP) shaft speed trip in Technical Specification (T.S.) 3.3.1, Table 3.3-2. The change is due to redefining the events which take place before the Control Element Assemblies drop into the core. During Cycle 1, the response time of .75 seconds was measured from the time a trip condition existed, such as a loss of power to the RCP motors, to the moment the Control Element Drive Mechanisms (CEDM) coil breakers opened. During Cycle 2 operation, the response time of .3 seconds will be defined from the time a signal is sent down the RCP shaft speed sensor line to the CPCs to the moment the CEDM coil breakers open.

B. PURPOSE OF THE TECHNICAL SPECIFICATION

The purpose of T.S. 3.3.1 is to ensure that (1) the associated Engineered Safety Features Actuation action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

C. NEED FOR THE TECHNICAL SPECIFICATION AMENDMENT

During the Cycle 1 startup testing, it was found that the projected Reactor Coolant flow rate trip software, housed in the Core Protection Calculators, which monitors the RCP shaft speed and projects what the Reactor Coolant System flow will be in the future, was too sensitive to small deviations in RCP shaft speeds and caused unnecessary trips to the Unit. To correct this problem, the software dealing with the projected flow rate trip was taken out. In its place, trip software, which trips the unit when the RCP shaft speed slows to 95% of its normal speed as did the projected flow rate trip, was installed. Because of this change, the response time, as defined for the RCP shaft speed trip, has been redefined for Cycle 2 to reflect the purpose of the new trip. As a result of the redefinition of the response time, the safety analysis for Cycle 2 has taken credit for the faster time and to ensure that the Unit is operated within the safety analysis, Table 3.3-2 will have to reflect the credited response time as it was used in the safety analysis.

D. BASIS FOR PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

1. The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in

accordance with a proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety.

A discussion of these standards as they relate to the amendment request follows:

Standard 1--Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated because the changed response time ensures sufficient margin for mitigating the most limiting Design Basis Event (DBE). The Cycle 2 safety analysis results are still bounded by the reference cycle analysis. Therefore, there is no increase in the probability or consequences of an accident previously evaluated.

Standard 2--Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated because the change maintains the margin of safety. The redefinition of the response time insures that the results of the Cycle 2 safety analysis will remain within the bounds of the Specified Acceptable Fuel Design Limits (SAFDLs) and, by maintaining the .3 second response time, the Unit will be operated within the realm of the safety analysis. Therefore, the change will not create the possibility of a new or different kind of accident.

Standard 3--Involve a significant reduction in a margin of safety.

The proposed change does not involve a significant reduction in a margin of safety because the change ensures the margin of safety for Cycle 2 is maintained. The analysis results show that there is sufficient margin to mitigate the most limiting DBE and that the results are bounded by the reference cycle. Therefore, no reduction in margin will arise.

2. The proposed amendment matches the guidance concerning the application of standards for determining whether or not a significant hazards consideration exists (51 FR 7751) by example:

   (iii) For a nuclear power reactor, a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptable criteria for the Technical Specifications, the analytical methods used to demonstrate conformance with the Technical Specifications and regulations are not significantly changed, and that NRC has previously found such methods acceptable.

E. SAFETY EVALUATION FOR THE AMENDMENT REQUEST

The proposed Technical Specification amendment will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR. The proposed change does not change or replace equipment or components which are important to safety. The change reflects the actual response time of the trip circuitry.

The proposed Technical Specification amendment will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR. The change maintains the margin of safety. The redefinition of the response time insures that the results of the Cycle 2 safety analysis will remain within the bounds of the Specified Acceptable Fuel Design Limits (SAFDLs) and, by maintaining the .3 second response time, the Unit will be operated within the realm of the safety analysis. This does not increase the possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR.

The proposed Technical Specification amendment will not reduce the margin of safety as defined in the basis for the Technical Specifications. The change ensures the margin of safety for Cycle 2 is maintained. The analysis results show that there is sufficient margin to mitigate the most limiting DBE and that the results are bounded by the reference cycle. Therefore, no reduction in margin will arise.

F. ENVIRONMENTAL IMPACT CONSIDERATION DETERMINATION

The proposed change request does not involve an unreviewed environmental question because operation of PVNGS Unit 2, in accordance with this change, would not:

1. Result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the staff's testimony to the Atomic Safety and Licensing Board; or

2. Result in a significant change in effluents or power levels; or

3. Result in matters not previously reviewed in the licensing basis for PVNGS which may have a significant environmental impact.

G. MARKED-UP TECHNICAL SPECIFICATION CHANGE PAGES

Limiting Conditions For Operation And Surveillance Requirements:

3/4 3-11

TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
I. TRIP GENERATION	
A. Process	
1. Pressurizer Pressure - High	≤ 1.15 seconds
2. Pressurizer Pressure - Low	≤ 1.15 seconds
3. Steam Generator Level - Low	≤ 1.15 seconds
4. Steam Generator Level - High	≤ 1.15 seconds
5. Steam Generator Pressure - Low	≤ 1.15 seconds
6. Containment Pressure - High	≤ 1.15 seconds
7. Reactor Coolant Flow - Low	≤ 0.58 second
8. Local Power Density - High	
a. Neutron Flux Power from Excore Neutron Detectors	≤ 0.75 second*
b. CEA Positions	≤ 1.35 second**
c. CEA Positions: CEAC Penalty Factor	≤ 0.75 second**
9. DNBR - Low	
a. Neutron Flux Power from Excore Neutron Detectors	≤ 0.75 second*
b. CEA Positions	≤ 1.35 second**
c. Cold Leg Temperature	≤ 0.75 second##
d. Hot Leg Temperature	≤ 0.75 second##
e. Primary Coolant Pump Shaft Speed	0.30 ~~≤ 0.75~~ second#
f. Reactor Coolant Pressure from Pressurizer	≤ 0.75 second###
g. CEA Positions: CEAC Penalty Factor	≤ 0.75 second**
B. Excore Neutron Flux	
1. Variable Overpower Trip	≤ 0.55 second*
2. Logarithmic Power Level - High	
a. Startup and Operating	≤ 0.55 second*
b. Shutdown	≤ 0.55 second*

ATTACHMENT 6

A. DESCRIPTION OF THE TECHNICAL SPECIFICATION AMENDMENT REQUEST

The proposed amendment revises the CEA Insertion Limits as set forth in Technical Specification (T.S.) 3.1.3.6. Operation of the regulating Control Element Assemblies (CEAs) during Cycle 2 will be more limited than in Cycle 1. The revisions to the curves will maintain the margin of safety and insure that there will be sufficient shutdown margin to handle the most limiting Anticipated Operational Occurrence (AOO) and limiting fault events.

B. PURPOSE OF THE TECHNICAL SPECIFICATION

The purpose of T.S. 3.1.3.6 is to ensure that (1) acceptable power distribution limits are maintained, (2) the minimum shutdown margin is maintained, and (3) the potential effects of CEA misalignments are limited to acceptable levels.

C. NEED FOR THE TECHNICAL SPECIFICATION AMENDMENT

The proposed changes made to the CEA Insertion Limits are due to the change in the Cycle 2 core physics. Because of the change to the core, the worth of the CEAs has changed and as a result, the effects of the dropped and ejected CEA events change. To ensure that there is sufficient margin to mitigate such events, CEA insertion has to be restricted by the insertion limits set forth in the proposed T.S. 3.1.3.6.

D. BASIS FOR PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

1. The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a proposed amendment would not: (1) Involve a significant increase in the probability of consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety.

   A discussion of these standards as they relate to the amendment request follows:

   Standard 1--Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated because by restricting the insertion of the rods to Gp 3 60" withdrawn, margin is maintained to mitigate the most limiting events, the dropped or ejected rod accidents as they are described in the FSAR. By complying with the proposed changes during Cycle 2 operation, the Cycle 2 safety analysis results will be bounded by the reference cycle (Cycle 1) safety analysis. This then ensures that the Cycle 2 operation will experience the same probability of consequences of an accident. The proposed change is made to ensure that Cycle 2 safety analysis is bounded by the reference cycle (Cycle 1) safety analysis. Therefore, there is no change in the probability or consequences of an accident occurring.

Standard 2--Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed change is more limiting than the reference cycle insertion limits. By restricting the insertion limits, there become fewer opportunities for the Unit to experience accidents. Since the change is more conservative a new or different kind of accident will not be created.

Standard 3--Involve a significant reduction in a margin of safety.

The proposed change does not involve a significant reduction in a margin of safety because the proposed change is being made to maintain Cycle 2 margin of safety and sufficient shutdown margin for the most limiting Anticipated Operational Occurrence (AOO) and limiting fault event. Therefore, the reduction of safety margin does not arise.

2. The proposed amendment matches the guidance concerning the application of standards for determining whether or not a significant hazards consideration exists (51 FR 7751) by example:

   (iii) For a nuclear power reactor, a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptable criteria for the Technical Specifications, the analytical methods used to demonstrate conformance with the Technical Specifications and regulations are not significantly changed, and that NRC has previously found such methods acceptable.

E. SAFETY EVALUATION FOR THE AMENDMENT REQUEST

The proposed Technical Specification amendment will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR. The proposed change is not a change or replace equipment or components important to safety. Therefore, there is no increase in the probability of occurrence or the consequences of an accident occurring.

The proposed Technical Specification amendment will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR. The proposed change places limits on the insertion of the CEAs such that the results from any accident occurring, while within the bounds set by T.S. Figure 3.1-3 and 3.1-4, will have the same consequences as those determined for the reference cycle. Thus, the proposed change is a result of maintaining the Cycle 2 safety analysis results within the reference cycle bounds and no new or different kinds of accidents will be created.

The proposed Technical Specification amendment will not reduce the margin of safety as defined in the basis for the Technical Specifications. The proposed change is being made to maintain Cycle 2 margin of safety and sufficient shutdown margin for the most limiting AOO and limiting fault events. Therefore, the reduction of safety margin does not arise.

F. ENVIRONMENTAL IMPACT CONSIDERATION DETERMINATION

The proposed change request does not involve an unreviewed environmental question because operation of PVNGS Unit 2, in accordance with this change, would not:

1. Result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the staff's testimony to the Atomic Safety and Licensing Board; or

2. Result in a significant change in effluents or power levels; or

3. Result in matters not previously reviewed in the licensing basis for PVNGS which may have a significant environmental impact.

G. MARKED-UP TECHNICAL SPECIFICATION CHANGE PAGES

Limiting Conditions For Operation And Surveillance Requirements:

3/4 1-31
3/4 1-32

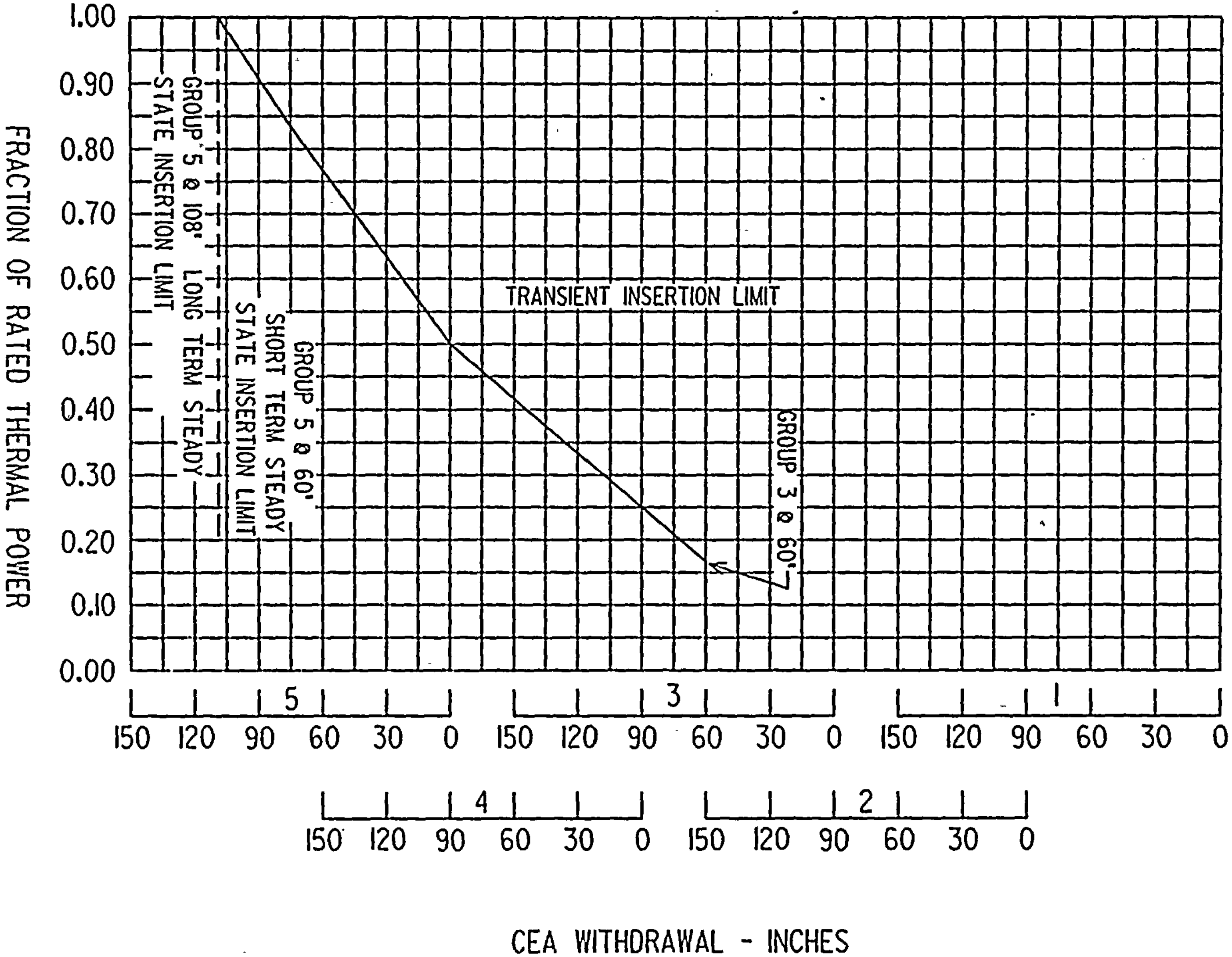

FIGURE 3.1-3

CEA INSERTION LIMITS VS THERMAL POWER

(COLSS IN SERVICE)

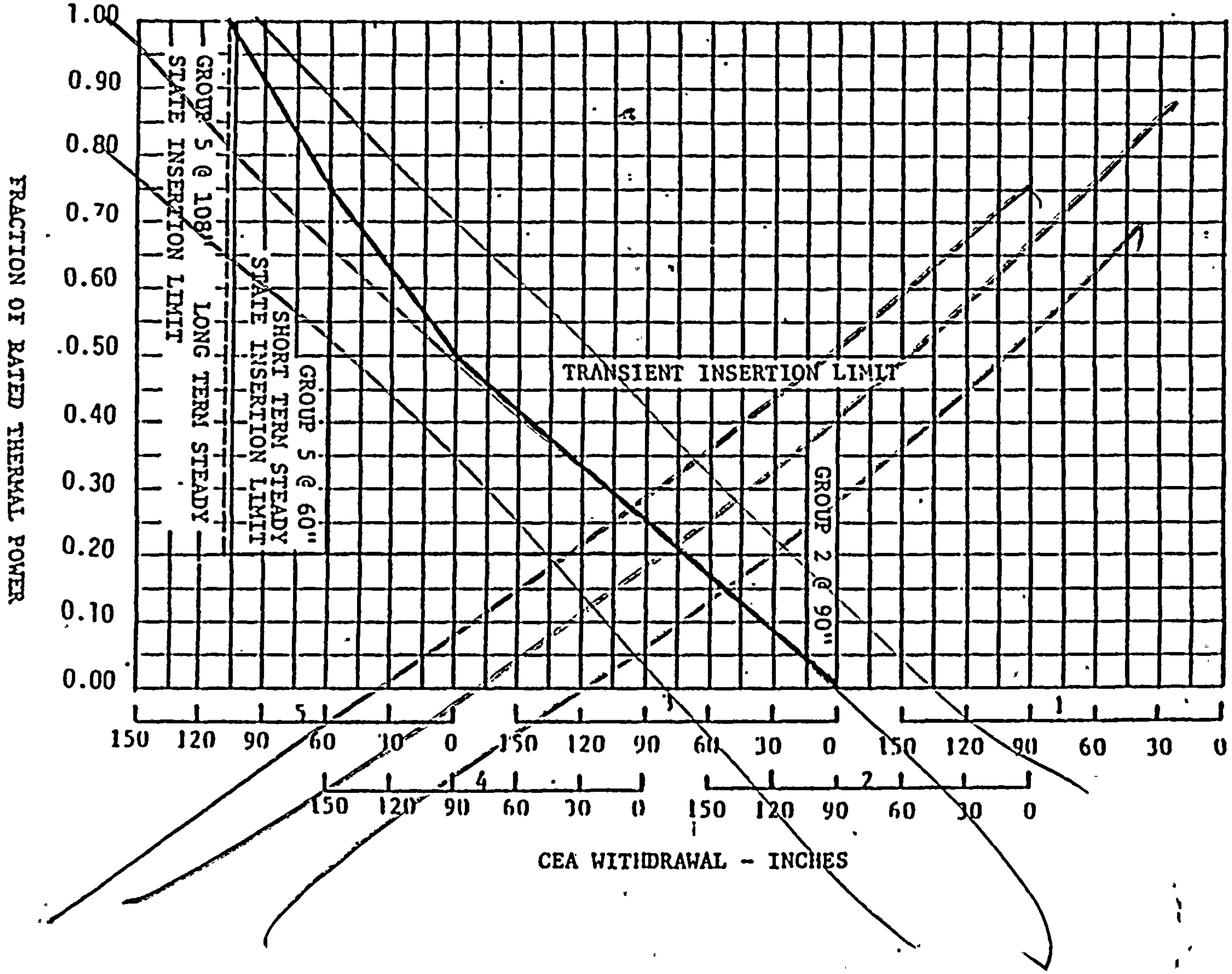

FIGURE 3.1-3

CEA INSERTION LIMITS VS THERMAL POWER (COLSS IN SERVICE)

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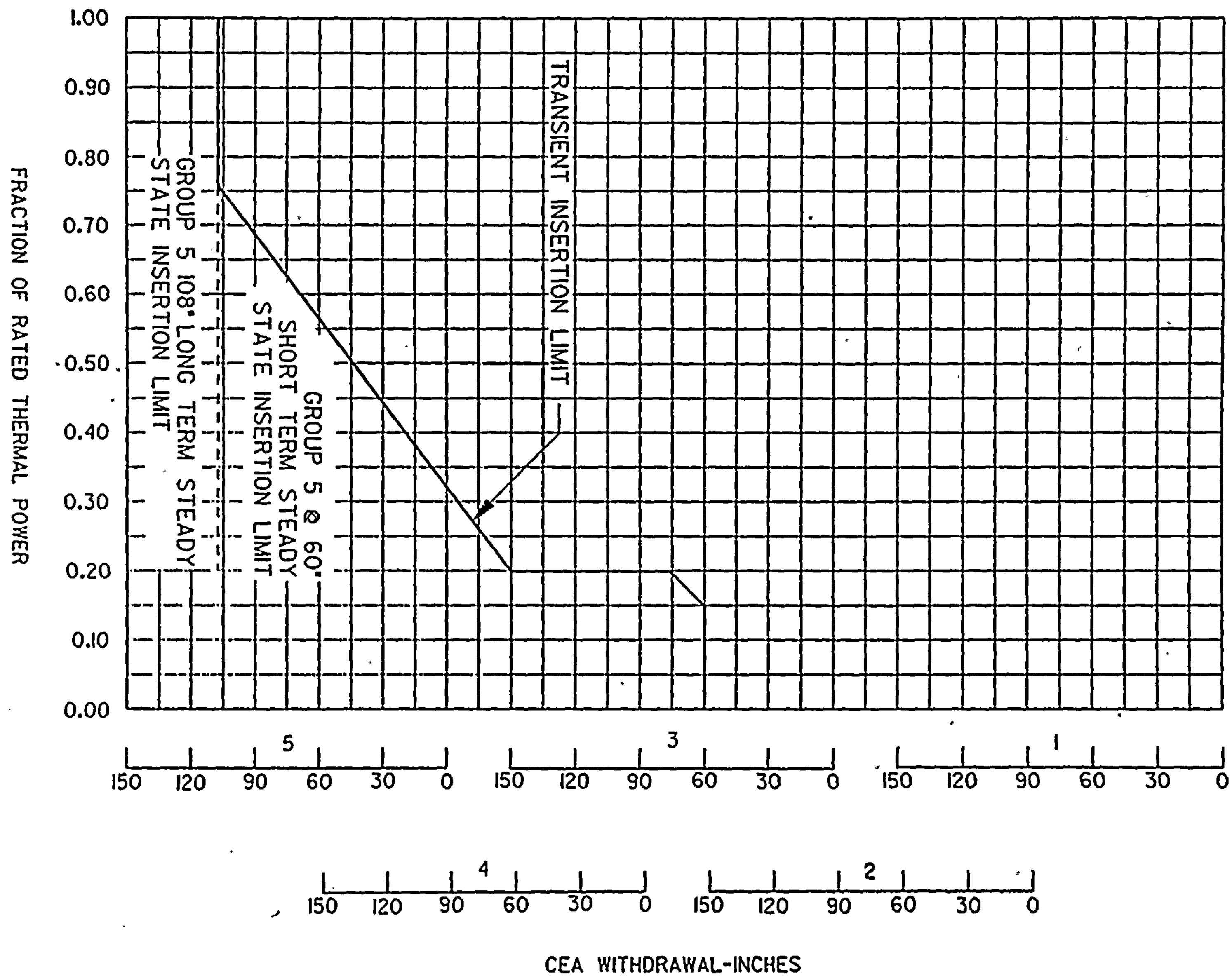

FIGURE 3.1-4

CEA INSERTION LIMITS VS THERMAL POWER

(COLSS OUT OF SERVICE)

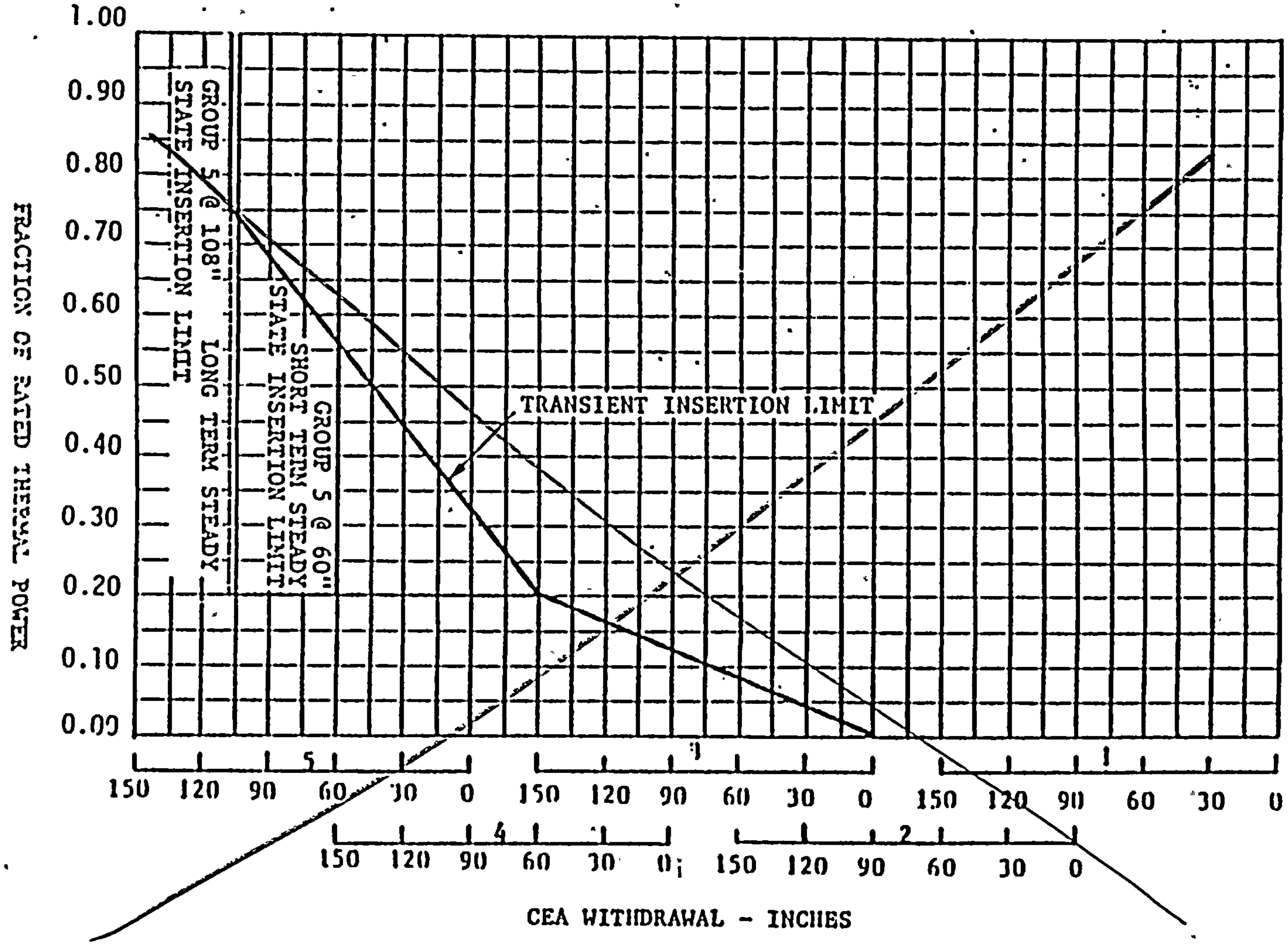

FIGURE 3.1-4

CEA INSERTION LIMITS VS THERMAL POWER (COLSS OUT OF SERVICE)

ATTACHMENT 7

A. DESCRIPTION OF THE PROPOSED CHANGE

The existing PVNGS Unit 1 Technical Specifications provide an allowance for entering penalty factors into the Core Protection Calculators (CPCs) to compensate for Resistance Temperature Detector (RTD) response times greater than 8 seconds (but less than or equal to 13 seconds). These CPC penalty factors are provided in Technical Specification Table 3.3-2a and are supported by the Cycle 1 safety analyses. However, the Cycle 2 safety analyses will not support these CPC penalty factors. Therefore, Table 3.3-2a must be deleted and Table 3.3-2 must be revised to remove this CPC penalty factor allowance.

B. PURPOSE OF THE TECHNICAL SPECIFICATION

Technical Specification Table 3.3-2 (and associated Table 3.3-2a) provide the allowable response times for instrumentation used in the PVNGS reactor protective system. By ensuring that the reactor protective instrumentation meets these response time requirements, the assumptions used in the safety analyses are complied with and the associated protective action (i.e., reactor trip) is received within the time frame allowed by the safety analyses.

The RTDs that are the subject of this proposed Technical Specification change measure the Reactor Coolant System (RCS) hot and cold leg temperatures. The temperature measurements are provided as an input to the CPCs for use in the DNBR calculation. Each CPC channel receives temperature inputs from both RCS hot legs and from two diametrically opposed RCS cold legs.

C NEED FOR THE TECHNICAL SPECIFICATION AMENDMENT

This Technical Specification change is necessary in order to ensure that the Cycle 2 safety analyses assumptions are complied with during Unit 1, Cycle 2 operations. The Cycle 2 safety analyses assume a maximum RTD response time of 8 seconds and do not include an allowance to enter CPC penalty factors to compensate for RTD response times greater than 8 seconds. Therefore, there should not be any allowances in the Technical Specifications for using the CPC penalty factors. For this reason, Technical Specification Table 3.3-2a should be deleted and Table 3.3-2 should be revised to remove the penalty factor allowances.

D. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION

1. The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a proposed amendment would not: (1) Involve a significant increase in the probability of consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety.

A discussion of these standards as they relate to the amendment request follows:

Standard 1 -- Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed Technical Specification change will not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change involves revising Table 3.3-2 and deleting Table 3.3-2a to remove the allowance which provides for CPC penalty factors to compensate for RTD response times greater than 8 seconds. The subject RTDs measure the RCS hot and cold leg temperatures and provide an input to the associated CPC channel for use in the CPC DNBR calculation. The response times of these RTDs has no impact on the probability of occurrence of any of the accidents that depend on a CPC low DNBR reactor trip.

This revision to Table 3.3-2 and the deletion of Table 3.3-2a will ensure that the consequences of the analyzed accidents will be no worse than evaluated for the Cycle 2 safety analyses. The existing Cycle 1 safety analyses support the use of CPC penalty factors to compensate for RTD response times slower than 8 seconds. The Cycle 2 safety analyses do not support the use of the CPC penalty factors. Thus, during Cycle 2, any RTD response times greater than 8 seconds will be unacceptable and the use of Table 3.3-2a will not be supported by the Cycle 2 safety analyses. Therefore, Table 3.3-2a should be deleted and Table 3.3-2 should be revised to assure that operation of PVNGS Unit 1 is in accordance with the Cycle 2 safety analyses.

Standard 2 -- Create the possibility of a new or different kind of accident from any accident previously analyzed.

This proposed Technical Specification change will not create the possibility of a new or different kind of accident from any accident previously analyzed. This proposed change, to delete the Technical Specification allowance for degraded RTD response times, does not affect the operation of the RTDs or the associated CPC channels. With the change, if a RTD response time is greater than 8 seconds, the associated CPC channel must be declared inoperable until repairs and/or retest are successfully completed.

Standard 3 -- Involve a significant reduction in a margin of safety.

This proposed Technical Specification change will not involve a significant reduction in a margin of safety. The basis for the existing Technical Specification Table 3.3-2a is the Cycle 1 safety analysis which analyzed the cases where the RTD response times were greater than 8 seconds but less than 13 seconds. For Cycle 2, there will not be an analysis to support the CPC penalty factors for degraded RTD response times. Therefore, Table 3.3-2a must be deleted since it will have no supporting basis during Cycle 2.

2. The Commission has provided guidance concerning the application of the Standards for determining whether a significant hazards consideration exists by providing certain examples (51 FR 7751) of amendments that are considered least likely to involve a significant hazards consideration. This proposed amendment matches example (ii) in that it is a change that constitutes an additional

limitation, restriction or control not presently included in the Technical Specifications. Specifically, this proposed Technical Specification change constitutes an additional limitation because the allowance for RTD response times greater than 8 seconds has been deleted. Thus, if a RTD response time is measured greater than 8 seconds, then that channel of the CPCs must be declared inoperable until repairs and/or retest are satisfactorily completed.

E. SAFETY EVALUATION FOR THE PROPOSED CHANGE

This proposed Technical Specification change will not increase the probability of occurrence of an accident previously evaluated in the FSAR. The subject RTDs measure the RCS hot and cold leg temperatures and provide an input to the CPCs for use in the CPC DNBR calculations. The response times of these RTDs have no effect on the probability of occurrence of any of the accidents that rely on a CPC low DNBR trip.

This proposed Technical Specification change will not increase the consequences of any accidents previously evaluated in the FSAR. The existing Cycle 1 safety analyses assure a RTD response time of no greater than 8 seconds. Additional analysis was performed for Cycle 1 to justify the application of CPC penalty factors if the measured RTD response times are greater than 8 seconds but no more than 13 seconds. This additional analysis supported the provisions contained in Technical Specification Tables 3.3-2 and 3.3-2a to apply CPC penalty factors to compensate for degraded RTD response times. The Cycle 2 safety analyses also assumed a maximum RTD response time of 8 seconds. However, no additional analysis was performed for Cycle 2 to support RTD response times greater than 8 seconds. Therefore, the Cycle 2 safety analyses do not support Table 3.3-2a and it must be deleted to ensure operation of PVNGS Unit 1 within the Cycle 2 safety analyses. Therefore, this Technical Specification change will ensure that the consequences of any accidents will be no greater than that of the Cycle 2 safety analyses.

This proposed Technical Specification change will not create the possibility of a new or different kind of accident from any accident previously evaluated. This proposed change, to delete the Technical Specifications allowance for degraded RTD response times, does not affect the operation of the RTDs or the associated CPC channels. With the change, if a RTD response time is greater than 8 seconds, the associated CPC channel must be declared inoperable until repairs and/or retest are successfully completed.

This Technical Specification change will not reduce the margin of safety as defined in the basis for any Technical Specifications. The basis for the existing Table 3.3-2a is the Cycle 1 safety analyses which analyzed the cases where the RTD response times were greater than 8 seconds but less than 13 seconds. For Cycle 2, there is no longer an analysis to support the CPC penalty factors for degraded RTD response times. Thus, Table 3.3-2a must be deleted since it will have no basis during Cycle 2.

[illegible]

F. ENVIRONMENTAL IMPACT CONSIDERATION DETERMINATION

The proposed change request does not involve an unreviewed environmental question because operation of PVNGS Unit 2, in accordance with this change, would not:

1. Result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the Staff's testimony to the Atomic Safety and Licensing Board (ASLB), Supplements to the FES, Environmental Impact Appraisals, or in any decisions of the ASLB; or

2. Result in matters not previously reviewed in the licensing basis for PVNGS which may have a significant environmental impact.

G. MARKED-UP TECHNICAL SPECIFICATION CHANGES PAGES

Enclosed are revised pages 3/4 3-12; 3/4 3-13 of the PVNGS Unit 2 Technical Specifications.

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
C. Core Protection Calculator System	
1. CEA Calculators	Not Applicable
2. Core Protection Calculators	Not Applicable
D. Supplementary Protection System	
Pressurizer Pressure - High	$\leq$ 1.15 second
II. RPS LOGIC	
A. Matrix Logic	Not Applicable
B. Initiation Logic	Not Applicable
III. RPS ACTUATION DEVICES	
A. Reactor Trip Breakers	Not Applicable
B. Manual Trip	Not Applicable

*Neutron detectors are exempt from response time testing. The ~~response time~~ RESPONSE TIME of the neutron flux signal portion of the channel shall be measured from the detector output or from the input of first electronic component in channel.

** ~~Response time~~ RESPONSE TIME shall be measured from the output of the sensor. Acceptable CEA sensor response shall be demonstrated by compliance with Specification 3.1.3.4.

#The pulse transmitters measuring pump speed are exempt from response time testing. The ~~response time~~ RESPONSE TIME shall be measured from the pulse shaper input.

~~Response time~~ RESPONSE TIME shall be measured from the output of the resistance temperature detector (sensor). RTD response time shall be measured at least once per 18 months. The measured response time of the slowest RTD shall be less than or equal to ~~13~~ 8 seconds. ~~Adjustments to the CPC addressable constants given in Table 3.3-2a shall be made to accommodate current values of the RTD time constants. If the RTD time constant for a CPC channel exceeds the value corresponding to the penalties currently in use, the affected channel(s) shall be declared inoperable until penalties appropriate to the new time constant are installed.~~

~~Response time~~ RESPONSE TIME shall be measured from the output of the pressure transmitter. The transmitter response time shall be less than or equal to 0.7 second.

TABLE 3.3-2a

INCREASES IN BERR0, BERR2, AND BERR4 VERSUS RTD DELAY TIMES

RTD DELAY TIME (τ)	BERR0 INCREASE (%)	BERR2 INCREASE (%)	BERR4 INCREASE (%)
$\tau \leq 8.0$ sec	0	0	0
$8.0 \text{ sec} < \tau \leq 10.0$ sec	2.5	2.0	1.0
$10.0 \text{ sec} < \tau \leq 13.0$ sec	6.0	4.0	6.0

NOTE: BERR term increases are not cumulative. For example, if the time constant changes from the range of $8.0 < \tau \leq 10.0$ sec to the range $10.0 < \tau \leq 13.0$, the BERR0 increase from its original ($\tau \leq 8.0$ sec) value is 6.0 not 2.5 + 6.0.

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ATTACHMENT 8

A. DESCRIPTION OF THE TECHNICAL SPECIFICATION AMENDMENT REQUEST

The proposed amendment changes references to the calculated Departure from Nucleate Boiling Ratio (DNBR) from 1.231 to 1.24 as set forth in Technical Specification (T.S) 2.1.1.1, Table 2.2-1, Basis 2.1.1, and Basis 2.2.1. The amendment also deletes references to the calculation of additional rod bow penalties if the rod bow penalty incorporated into the DNBR limit is not sufficient for any part of the cycle. The low pressurizer pressure floor is also changed from 1861 to 1860 because of the changed DNBR value.

B. PURPOSE OF THE TECHNICAL SPECIFICATION

The purpose of T.S. 2.1.1 is to 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 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.

C. NEED FOR THE TECHNICAL SPECIFICATION AMENDMENT

During Cycle 1 operation, the rod bow penalty factor was applied to the DNBR in increments. This method provided a means for not penalizing the operational margin unnecessarily during the cycle. As the fuel assemblies approach higher burnup the advantage of the Cycle 1 method no longer exists. The application of a rod bow penalty factor large enough to provide protection throughout the cycle is now more advantageous. This can be accomplished because the physics of the Cycle 2 core is such that, by applying a rod bow penalty factor of 1.75% Minimum DNBR (MDNBR) to the DNBR limit, there will be sufficient margin to compensate for the effects of rod bow caused by those bundles with burnups of less than 30,000 MWD/MTU. For those bundles with burnups of greater than 30 GWD/MTU, there is sufficient margin from other factors to offset the small increase in the rod bow penalty.

As a result of the DNBR change, a reevaluation of the safety analysis was performed to determine if the low pressurizer pressure floor for the DNBR-low trip would change. The low DNBR trip provides protection in the event of an increase in heat removal by the secondary system and subsequent cooldown of the reactor coolant. The analysis has shown that a pressurizer pressure of 1860 instead of 1861 will ensure that, if a reactor trip occurs on Low-DNBR, the plant will not reach the Specified Acceptable Fuel Design Limits (SAFDLs).

D BASIS FOR PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

1. The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility in accordance with a proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety.

   A discussion of these standards as they relate to the amendment request follows:

   Standard 1--Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed change incorporates the reference cycle (Cycle 1) approved fuel rod bow penalty factor into the DNBR limit for fuel assembly burnups of up to 30,000 MWD/MTU. For those assemblies which will reach burnups of greater than 30,000 MWD/MTU in Cycle 2, there is sufficient available margin, due to lower radial power peaks, to offset any increase in the rod bow penalty. Thus, the probability or consequences of an accident occurring during Cycle 2 is the same as the reference cycle.

   Standard 2--Create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed change incorporates the reference cycle approved fuel rod bow penalty factor into the DNBR limit for fuel assembly burnups of up to 30,000 MWD/MTU. For those assemblies which will reach burnups of greater than 30,000 MWD/MTU in Cycle 2, there is sufficient available margin, due to lower radial power peaks, to offset any increase in the rod bow penalty. Therefore, the possibility of a new or different kind of accident will not increase.

   Standard 3--Involve a significant reduction in a margin of safety.

   The proposed change does not involve a significant reduction in a margin of safety because the proposed change incorporates the reference cycle approved fuel rod bow penalty factor in the DNBR limit for fuel assembly burnups of up to 30,000 MWD/MTU. For those assemblies which will reach burnups of greater than 30,000 MWD/MTU in Cycle 2, there is sufficient available margin, due to lower radial power peaks, to offset any increase in the rod bow penalty. Therefore, there is no reduction in the margin of safety.

2. The proposed amendment matches the guidance concerning the application of standards for determining whether or not a significant hazards consideration exists (51 FR 7751) by example:

   (iii) For a nuclear power reactor, a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptable criteria for the Technical Specifications, the analytical methods used to demonstrate conformance with the Technical Specifications and regulations are not significantly changed, and that NRC has previously found such methods acceptable.

E. SAFETY EVALUATION FOR THE AMENDMENT REQUEST

The proposed Technical Specification amendment will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR. The proposed change does not change or replace any equipment or components important to safety. The proposed change changes the DNBR margin by incorporating the reference cycle approved fuel rod bow penalty for a burnup of up to 30,000 MWD/MTU. Assemblies which will reach a burnup of greater than 30,000 MWD/MTU in Cycle 2, will not contribute a large enough rod bow penalty to require a larger penalty factor to be applied to the DNBR limit. The reference cycle safety analysis has incorporated into the analysis results. The effects of the higher burnups and, therefore, the DNBR for Cycle 2 is bounded by the reference cycle.

The proposed Technical Specification amendment will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR. The proposed change is bounded by the reference cycle safety analysis because the effects of higher burnups on the fuel rod bow penalty factor were incorporated into the analysis. Therefore, the possibility of a new or different kind of accident stays the same.

The proposed Technical Specification amendment will not reduce the margin of safety as defined in the basis for the technical specifications. The proposed change is bounded by the reference cycle safety analysis because the effects of higher burnups on the fuel rod bow penalty factor were incorporated into the analysis. Therefore, the margin of safety stays the same.

F. ENVIRONMENTAL IMPACT CONSIDERATION DETERMINATION

The proposed change request does not involve an unreviewed environmental question, because operation of PVNGS Unit 2, in accordance with this change, would not:

1. Result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the staff's testimony to the Atomic Safety and Licensing Board; or

2. Result in a significant change in effluents or power levels; or

3. Result in matters not previously reviewed in the licensing basis for PVNGS which may have a significant environmental impact.

G. MARKED-UP TECHNICAL SPECIFICATION CHANGE PAGES

Limiting Conditions For Operation And Surveillance Requirements:

2-1	B 2-5
2-3	B 2-6
2-5	
B 2-1	
B 2-2	

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

2.1.1 REACTOR CORE

DNBR

2.1.1.1 The calculated DNBR of the reactor core shall be maintained greater than or equal to ~~1.231~~.1.24

APPLICABILITY: MODES 1 and 2.

ACTION:

1.24
Whenever the calculated DNBR of the reactor has decreased to less than ~~1.231~~, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

PEAK LINEAR HEAT RATE

2.1.1.2 The peak linear heat rate (adjusted for fuel rod dynamics) of the fuel shall be maintained less than or equal to 21 kW/ft.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the peak linear heat rate (adjusted for fuel rod dynamics) of the fuel has exceeded 21 kW/ft, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4, and 5:

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

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TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
I. TRIP GENERATION		
A. Process		
1. Pressurizer Pressure - High	≤ 2383 psia	≤ 2388 psia
2. Pressurizer Pressure - Low	≥ 1837 psia (2)	≥ 1822 psia (2)
3. Steam Generator Level - Low	≥ 44.2% (4)	≥ 43.7% (4)
4. Steam Generator Level - High	≤ 91.0% (9)	≤ 91.5% (9)
5. Steam Generator Pressure - Low	≥ 919 psia (3)	≥ 912 psia (3)
6. Containment Pressure - High	≤ 3.0 psig	≤ 3.2 psig
7. Reactor Coolant Flow - Low		
a. Rate	≤ 0.115 psi/sec (6)(7)	≤ 0.118 psi/sec (6)(7)
b. Floor	≥ 11.9 psid (6)(7)	≥ 11.7 psid(6)(7)
c. Band	≤ 10.0 psid (6)(7)	≤ 10.2 psid (6)(7)
8. Local Power Density - High	≤ 21.0 kW/ft (5)	≤ 21.0 kW/ft (5)
9. DNBR - Low	≥ ~~1.231~~ 1.24 (5)	≥ ~~1.231~~ 1.24 (5)
B. Excore Neutron Flux		
1. Variable Overpower Trip		
a. Rate	≤ 10.6%/min of RATED THERMAL POWER (8)	≤ 11.0%/min of RATED THERMAL POWER (8)
b. Ceiling	≤ 110.0% of RATED THERMAL POWER (8)	≤ 111.0% of RATED THERMAL POWER (8)
c. Band	≤ 9.8% of RATED THERMAL POWER (8)	≤ 10.0% of RATED THERMAL POWER (8)

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATIONS

(1) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(2) In MODES 3-4, value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(3) In MODES 3-4, value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower level wide range instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties. ~~and dynamic allowances.~~ Trip may be manually bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 1% of RATED THERMAL POWER.

DELETED

The approved DNBR limit is 1.231 which includes a partial rod bow penalty compensation. If the fuel burnup exceeds that for which an increased rod bow penalty is required, the DNBR limit shall be adjusted. In this case a DNBR trip setpoint of 1.231 is allowed provided that the difference is compensated by an increase in the CPC addressable constant BERR1 as follows:

$$BERR1_{new} = BERR1_{old} \left[1 + \frac{RB - RB_o}{100} \times \frac{d\,(\%\ POL)}{d\,(\%\ DNBR)}\right]$$

where $BERR1_{old}$ is the uncompensated value of BERR1; RB is the fuel rod bow penalty in % DNBR; RB_o is the fuel rod bow penalty in % DNBR already accounted for in the DNBR limit; POL is the power operating limit; and d (% POL)/d (% DNBR) is the absolute value of the most adverse derivative of POL with respect to DNBR.

2.1.1 REACTOR CORE

The restrictions of these safety limits 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 cladding is prevented by (1) 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, and (2) maintaining the dynamically adjusted peak linear heat rate of the fuel at or less than 21 kW/ft which will not cause fuel centerline melting in any fuel rod.

First, by operating within the nucleate boiling regime of heat transfer, the heat transfer coefficient is large enough so that the maximum clad surface temperature is only slightly greater than the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in higher cladding temperatures and the possibility of cladding failure.

Correlations predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the predicted DNB heat flux at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB. The minimum value of DNBR during normal operation and design basis anticipated operational occurrences is limited to ~~1.231~~ 1.24 based upon a statistical combination of CE-1 CHF correlation and engineering factor uncertainties and is established as a Safety Limit. The DNBR limit of ~~1.231~~ 1.24 includes a rod bow compensation of 1.75% ~~0.8%~~ on DNBR. ~~For fuel burnups which exceed that for which an increased rod bow penalty is required, the DNBR limit shall be adjusted. In this case the DNBR trip setpoint of 1.231 is allowed if the required DNBR increase is compensated by an increase of the addressable constant BERR1.~~ Deleted

Second, operation with a peak linear heat rate below that which would cause fuel centerline melting maintains fuel rod and cladding integrity. Above this peak linear heat rate level (i.e., with some melting in the center), fuel rod integrity would be maintained only if the design and operating conditions are appropriate throughout the life of the fuel rods. Volume changes which accompany the solid to liquid phase change are significant and require accommodation. Another consideration involves the redistribution of the fuel which depends on the extent of the melting and the physical state of the fuel rod at the time of melting. Because of the above factors, the steady state value of the peak linear heat rate which would not cause fuel centerline melting is established as a Safety Limit. To account for fuel rod dynamics (lags), the directly indicated linear heat rate is dynamically adjusted by the CPC program.

Limiting Safety System Settings for the Low DNBR, High Local Power Density, High Logarithmic Power Level, Low Pressurizer Pressure and High Linear Power Level trips, and Limiting Conditions for Operation on DNBR and kW/ft margin are specified such that there is a high degree of confidence that the specified acceptable fuel design limits are not exceeded during normal operation and design basis anticipated operational occurrences.

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 Coolant System components are designed to Section III, 1974 Edition, Summer 1975 Addendum, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of 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.

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and Reactor Coolant System are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

The DNBR - Low and Local Power Density - High are digitally generated trip setpoints based on Safety Limits of ~~1.231~~ 1.24 and 21 kW/ft, respectively. Since these trips are digitally generated by the Core Protection Calculators, the trip values are not subject to drifts common to trips generated by analog type equipment. The Allowable Values for these trips are therefore the same as the Trip Setpoints.

To maintain the margins of safety assumed in the safety analyses, the calculations of the trip variables for the DNBR - Low and Local Power Density - High trips include the measurement, calculational and processor uncertainties and dynamic allowances as defined in CESSAR System 80 applicable system descriptions and safety analyses.

Local Power Density - High (Continued)

a. Nuclear flux power and axial power distribution from the excore flux monitoring system;

b. Radial peaking factors from the position measurement for the CEAs;

c. Delta T power from reactor coolant temperatures and coolant flow measurements.

The local power density (LPD), the trip variable, calculated by the CPC incorporates uncertainties and dynamic compensation routines. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core peak LPD is sufficiently less than the fuel design limit such that the increase in actual core peak LPD after the trip will not result in a violation of the Peak Linear Heat Rate Safety Limit. CPC uncertainties related to peak LPD are the same types used for DNBR calculation. Dynamic compensation for peak LPD is provided for the effects of core fuel centerline temperature delays (relative to changes in power density), sensor time delays, and protection system equipment time delays.

DNBR - Low

The DNBR - Low trip is provided to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit in the event of design bases anticipated operational occurrences. The DNBR - Low trip incorporates a low pressurizer pressure floor of ~~1861~~ 1840 psia. At this pressure a DNBR - Low trip will automatically occur. The DNBR is calculated in the CPC utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore neutron flux monitoring system;

b. Reactor Coolant System pressure from pressurizer pressure measurement;

c. Differential temperature (Delta T) power from reactor coolant temperature and coolant flow measurements;

d. Radial peaking factors from the position measurement for the CEAs;

e. Reactor coolant mass flow rate from reactor coolant pump speed;

f. Core inlet temperature from reactor coolant cold leg temperature measurements.

DNBR - Low (Continued)

The DNBR, the trip variable, calculated by the CPC incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the calculated core DNBR is sufficiently greater than ~~1.231~~ 1.24 such that the decrease in calculated core DNBR after the trip will not result in a violation of the DNBR Safety Limit. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modelling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.

The DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.

	Parameter	Limiting Value
a.	RCS Cold Leg Temperature-Low	≥ 470°F
b.	RCS Cold Leg Temperature-High	≤ 610°F
c.	Axial Shape Index-Positive	Not more positive than + 0.5
d.	Axial Shape Index-Negative	Not more negative than - 0.5
e.	Pressurizer Pressure-Low	≥ ~~1861~~ 1840 psia
f.	Pressurizer Pressure-High	≤ 2388 psia
g.	Integrated Radial Peaking Factor-Low	≥ 1.28
h.	Integrated Radial Peaking Factor-High	≤ 4.28
i.	Quality Margin-Low	≥ 0

Steam Generator Level - High

The Steam Generator Level - High trip is provided to protect the turbine from excessive moisture carry over. Since the turbine is automatically tripped when the reactor is tripped, this trip provides a reliable means for providing protection to the turbine from excesssive moisture carryover. This trip's setpoint does not correspond to a safety limit, and provides protection in the event of excess feedwater flow. The setpoint is identical to the main steam isolation setpoint. Its functional capability at the specified trip setting enhances the overall reliability of the reactor protection system.

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Attachment 9

A. DESCRIPTION OF THE TECHNICAL SPECIFICATION AMENDMENT REQUEST

The proposed amendment changes the Reactor Coolant System (RCS) total flow rate as set forth in Technical Specification (T.S.) 3.2.5 from greater than or equal to 164.0 x 10^6 lbm/hr to greater than or equal to 155.8 x 10^6 lbm/hr.

B. PURPOSE OF THE TECHNICAL SPECIFICATION

The purpose of T.S. 3.2.5 ensures that the actual RCS total flow rate is maintained at or above the minimum value used in the safety analysis.

C. NEED FOR THE TECHNICAL SPECIFICATION AMENDMENT

T.S. 3.2.5 is being changed to eliminate an ambiguity in where instrument uncertainty is to be included when comparing measured RCS flow rate against the RCS flow rate used in the safety analysis. As currently worded, actual total RCS flow rate is to be compared against the 100% design flow value of 164.0 x 10^6 lbm/hr. The term "actual" implies that the RCS flow rate determined by the Reactor Coolant Pump (RCP) delta-pressure method is to be corrected for pressure transmitter uncertainty. The uncertainty amounts to a maximum of 4% of flow for transmitters within their calibration period. The corrected flow rate is then compared to 164.0 x 10^6 lbm/hr. The RCS flow rate used in the safety analysis, however, is 95% of the design flow or 155.8 x 10^6 lbm/hr. The 100% design flow rate of 164.0 x 10^6 lbm/hr conservatively accommodated the maximum instrument uncertainty of 4%, removing the need to correct for instrument uncertainty. The T.S. basis states that the specification is provided to ensure that the actual total RCS flow rate is maintained at or above the minimum value used in the safety analysis. This T.S. change will remove the ambiguity and permit any changes in instrument uncertainty to be handled procedurally rather than requiring additional T.S. changes.

D. BASIS FOR PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

1. The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety.

   A discussion of these standards as they relate to the amendment request follows:

Standard 1--Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated because the value of 155.8 x 10^6 lbm/hr for minimum RCS flow rate is the value used in the reference cycle (Cycle 1) safety analysis. Therefore, the probability or consequences of an accident is the same for Cycle 2 as it is for the reference cycle.

Standard 2--Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated because the same value was used for both the reference cycle and Cycle 2 safety analysis. Therefore there is no possibility of creating a new or different kind of accident with the reduced RCS total flow.

Standard 3--Involve a significant reduction in a margin of safety.

The proposed change does not involve a significant reduction in the margin of safety because no changes have been made to the safety analysis. The proposed value in the T.S. is the value used in both the reference cycle and Cycle 2 safety analysis. Therefore, the margin of safety is the same for Cycle 2 as it is for the reference cycle.

2. The proposed amendment matches the guidance concerning the application of standards for determining whether or not a significant hazards consideration exists (51 FR 7751) by example:

(iii) For a nuclear power reactor, a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptable criteria for the technical specifications, the analytical methods used to demonstrate conformance with the technical specifications and regulations are not significantly changed, and that NRC has previously found such methods acceptable.

E. SAFETY EVALUATION FOR THE AMENDMENT REQUEST

The proposed Technical Specification amendment will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR. The proposed change does not change or replace equipment or components important to safety. The safety analysis for the proposed change is the same as the reference cycle and, therefore, the probability of occurrence or the consequences of an accident is the same.

The proposed technical specification amendment will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR. The Cycle 2 safety analysis for the proposed change uses the same value for RCS minimum flowrate as for the reference cycle and therefore, the possibility for an accident is the same.

The proposed Technical Specification amendment will not reduce the margin of safety as defined in the bases for the technical specifications. No changes have been made to the safety analysis. The proposed value in the T.S. is the value used in both the reference cycle and Cycle 2 safety analysis. Therefore, there is no reduction in the margin of safety.

F. ENVIRONMENTAL IMPACT CONSIDERATION DETERMINATION

The proposed change request does not involve an unreviewed environmental question because operation of PVNGS Unit 2, in accordance with this change, would not:

1. Result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the staff's testimony to the Atomic Safety and Licensing Board; or

2. Result in a significant change in effluents or power levels; or

3. Result in matters not previously reviewed in the licensing basis for PVNGS which may have a significant environmental impact.

G. MARKED-UP TECHNICAL SPECIFICATION CHANGE PAGES

Limiting Condition for Operation and Surveillance Requirements:

3/4 2-8
B 3/4 2-4

3/4.2.5 RCS FLOW RATE

LIMITING CONDITION FOR OPERATION

3.2.5 The actual Reactor Coolant System total flow rate shall be greater than or equal to ~~164.0×10^6~~ 155.8×10^6 lbm/hr.

APPLICABILITY: MODE 1.

ACTION:

With the actual Reactor Coolant System total flow rate determined to be less than the above limit, reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 The actual Reactor Coolant System total flow rate shall be determined to be greater than or equal to its limit at least once per 12 hours.

CONTROLLED BY USER

BASES

3/4.2.5 RCS FLOW RATE

This specification is provided to ensure that the actual RCS total flow rate is maintained at or above the minimum value used in the safety analyses.

add

3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

This specification is provided to ensure that the actual value of reactor coolant cold leg temperature is maintained within the range of values used in the safety analyses.

3/4.2.7 AXIAL SHAPE INDEX

This specification is provided to ensure that the actual value of the core average AXIAL SHAPE INDEX is maintained within the range of values used in the safety analyses.

3/4.2.8 PRESSURIZER PRESSURE

This specification is provided to ensure that the actual value of pressurizer pressure is maintained within the range of values used in the safety analyses.

The minimum value used in the safety analysis is 95% of the design flow rate (164.0×10^6 lbm/hr) or 155.8×10^6 lbm/hr. The actual RCS flowrate is determindd by direct measurement and an uncertainty associated with that measurement is considered when comparing actual RCS flow rate to the minimum required value of 155.8×10^6 lbm/hr.

ATTACHMENT 10

A. DESCRIPTION OF THE TECHNICAL SPECIFICATION AMENDMENT REQUEST

The proposed amendment changes the Linear Heat Rate (LHR) limit as defined in Technical Specification (T.S.) 3.2.1 from 14.0 kw/ft to 13.5 kw/ft. The change also provides information for the appropriate methods of monitoring LHR and formats the T.S. with regard to human factors.

B. PURPOSE OF THE TECHNICAL SPECIFICATION

The purpose of T.S. 3.2.1 is to limit Linear Heat Rate which will ensure that, in the event of a Loss of Coolant Accident (LOCA), the peak temperature of the fuel cladding will not exceed 2200°F.

C. NEED FOR THE TECHNICAL SPECIFICATION AMENDMENT

In support of the Unit 1 reload, the reanalysis of the Safety Analyses resulted in a change in the Linear Heat Rate limit to ensure the peak fuel clad temperature is not exceeded. The change in the LHR is, in part, due to the change in the method of performing the safety analysis. As part of the analysis, penalties are applied to compensate for increased power peaking caused by the densification of small interpellet gaps. These penalties are called Augmentation Factors and were not used for the Cycle 2 analysis. This method change has been approved by the NRC in "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 104 to Facility Operating License No. DPR-53, Baltimore Gas and Electric Company, Calvert Cliffs Nuclear Power Plant Unit No. 1, Docket No. 50-317". Other factors contributing to the change in LHR are from increased fuel enrichment and the core loading pattern.

In addition to changing the references to LHR, the amendment also delineates how LHR is to be monitored. By providing more detail of the monitoring of LHR, assurance is provided that the LHR will be maintained below the specified limit. The amendment also changes the format of the ACTION statement in such a way as to facilitate assessment of the actions required if the limit should be exceeded.

D. BASIS FOR PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

1. The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety.

A discussion of these standards as they relate to the amendment request follows:

Standard 1--Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated because the safety analysis of the proposed change is bounded by the safety limits set forth by 10 CFR 50.46. Changing the LHR limit will ensure that there is sufficient margin for the most limiting Design Basis Event (DBE). The change is also more conservative than the value used in Cycle 1. The format changes to the LCO and Action statements further define and clarify the actions required to be taken to ensure maintaining the LHR below the limit. Therefore, there will be no increase in the probability or consequences of an accident.

Standard 2--Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated because the safety analysis results of the proposed change are bounded by the safety limits set forth by 10 CFR 50.46. The proposed change to the LHR is more conservative than the LHR allowed by Cycle 1, thus reducing the consequences of an event but not creating any new or different accidents. The format modification changes the presentation of information within the T.S. but does not delete required actions and adds additional restrictions. Therefore, there will be no increase in the possibility of a new or different kind of accident.

Standard 3--Involve a significant reduction in a margin of safety.

The proposed change does not involve a significant reduction in a margin of safety because the safety analysis results of the proposed change are bounded by the safety limits set forth by 10 CFR 50.46. Changing the LHR limit will maintain sufficient margin for the most limiting DBE. Therefore, there will be no reduction in the safety margin.

2. The proposed amendment matches the guidance concerning the application of standards for determining whether or not a significant hazards consideration exists (51 FR 7751) by examples:

   (i) A purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error or a change in nomenclature.

and

(iii) For a nuclear power reactor, a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptable criteria for the Technical Specifications, the analytical methods used to demonstrate conformance with the Technical Specifications and regulations are not significantly changed, and that NRC has previously found such methods acceptable.

E. SAFETY EVALUATION FOR THE AMENDMENT REQUEST

The proposed Technical Specification amendment will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR. The safety analysis results of the proposed change are bounded by the safety limits set forth by 10 CFR 50.46 and do not change or replace equipment or components which are important to safety.

The proposed Technical Specification amendment will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR. The safety analysis results of the proposed change are bounded by the safety limits set forth by 10 CFR 50.46. The proposed change to the LHR is more conservative than the LHR allowed by the reference cycle (Cycle 1), thus reducing the consequences of an event but not creating any new or different accident or malfunction. The format modification changes the presentation of information within the T.S., but does not delete required actions and adds additional restrictions.

The proposed Technical Specification amendment will not reduce the margin of safety as defined in the basis for the technical specifications. The safety analysis results of the proposed change are bounded by the safety limits set forth by 10 CFR 50.46. Changing the LHR limit for Cycle 2 will maintain sufficient margin for the most limiting DBE, thus maintaining the margin of safety.

F. ENVIRONMENTAL IMPACT CONSIDERATION DETERMINATION

The proposed change request does not involve an unreviewed environmental question because operation of PVNGS Unit 2, in accordance with this change, would not:

1. Result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the staff's testimony to the Atomic Safety and Licensing Board; or

2. Result in a significant change in effluents or power levels; or

3. Result in matters not previously reviewed in the licensing basis for PVNGS which may have a significant environmental impact.

G. MARKED-UP TECHNICAL SPECIFICATION CHANGE PAGES

Limiting Conditions For Operation And Surveillance Requirements:

3/4 2-1
B 3/4 2-1

3/4.2 POWER DISTRIBUTION LIMITS

3/4 2.1 LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate ~~shall not exceed 14.0~~ limit of 13.5 kW/ft shall be maintained by one of the following methods as applicable:

a. Maintaining COLSS calculated core power less than or equal to the COLSS calculated power operating limit based on linear heat rate (when COLSS is in service); or

b. Maintaining peak linear heat rate within its limit using any operable CPC channel (when COLSS is out of service).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With the linear heat rate ~~exceeding its limits~~ limit not being maintained, as indicated by: ~~either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on kW/ft; or (2) when the COLSS is not being used, any OPERABLE Local Power Density channel exceeding the linear heat rate limit, within~~

1. COLSS calculated core power exceeding the COLSS calculated core power operating limit based on linear heat rate; or

2. Peak linear heat rate outside its limit using any operable CPC channel (when COLSS is out of service);

within 15 minutes initiate corrective action to reduce the linear heat rate to within the limits and either:

a. Restore the linear heat rate to within its limits within 1 hour, or

b. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limit~~s~~ when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the linear heat rate, as indicated on ~~all~~ Any OPERABLE Local Power Density channel~~s~~, is ~~less than or equal to 14.0 kW/ft.~~ within its limit.

4.2.1.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on ~~14.0 kW/ft.~~ linear heat rate.

Palo Verde Unit 2

BASES

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 Core Operating Limit Supervisory System (COLSS) and the Local Power Density channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core power operating limit corresponding to the allowable peak linear heat rate. Reactor operation at or below this calculated power level assures that the limits of ~~14.0~~ 13.5 kW/ft are not exceeded.

The COLSS calculated core power and the COLSS calculated core power operating limits based on linear heat rate are continuously monitored and displayed to the operator. A COLSS alarm is annunciated in the event that the core power exceeds the core power operating limit. This provides adequate margin to the linear heat rate operating limit for normal steady-state operation. Normal reactor power transients or equipment failures which do not require a reactor trip may result in this core power operating limit being exceeded. In the event this occurs, COLSS alarms will be annunciated. If the event which causes the COLSS limit to be exceeded results in conditions which approach the core safety limits, a reactor trip will be initiated by the Reactor Protective Instrumentation. The COLSS calculation of the linear heat rate includes appropriate penalty factors which provide, with a 95/95 probability/confidence level, that the maximum linear heat rate calculated by COLSS is conservative with respect to the actual maximum linear heat rate existing in the core. These penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering heat flux uncertainty, axial densification, software algorithm modelling, computer processing, rod bow, and core power measurement.

Parameters required to maintain the operating limit power level based on linear heat rate, margin to DNB, and total core power are also monitored by the CPCs ~~(assuming minimum core power of 20% of RATED THERMAL POWER). The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings~~. Therefore, in the event that the COLSS is not being used, operation within the linear heat rate limit~~s~~ ~~of Figure 3.2-2~~ can be maintained by utilizing any operable ~~a predetermined local power density margin and a total core power limit in the~~ CPC ~~trip~~ channel~~s~~. The above listed uncertainty and penalty factors plus those associated with the CPC startup test acceptance criteria are also included in the CPCs.

ATTACHMENT 11

A. DESCRIPTION OF THE TECHNICAL SPECIFICATION AMENDMENT REQUEST

The proposed amendment will revise Technical Specifications (T.S) 3.2.4, 3.3.1, Bases 3.1.3.1/3.1.3.2 and Bases 3.2.4. The changes are as follows:

T.S. 3.2.4-(1) Replaces the T.S. with a new format which addresses the specific conditions for monitoring DNBR with or without COLSS and/or the CEACs, (2) delineates by a new format what ACTIONS should be taken, (3) removes reference to the DNBR Penalty Factor table used in T.S. 4.2.4.4, and (4) replaces the present graph figures 3.2-1 and 3.2-2 of the DNBR limits with graph figures 3.2-1, 3.2-2 and 3.2-2a addressing DNBR operating limits for the conditions mentioned in (1) above.

T.S. 3.3.1-(1) Removes references to the operation of the reactor with both CEACs inoperable and with or without COLSS inservice, and (2) deletes the graph of DNBR margin operating limit, Figure 3.3-1, based on COLSS for both CEACs inoperable. These changes are a result of being incorporated into the proposed T.S. 3.2.4

Bases 3.1.3.1/3.1.3.2-(1) Removes references to Cycle 1 specific information, and (2) modifies Bases due to T.S. 3.2.4 changes.

Bases 3.2.4-Modifies Bases due to the T.S. 3.2.4 changes.

These changes are due, in part, to ensuring operation of Cycle 2 within the approved safety analysis and to improving the Technical Specifications from a human factors point of view.

B. PURPOSE OF THE TECHNICAL SPECIFICATION

The purpose of T.S. 3.2.4 is to ensure the limitation of DNBR, as a function of AXIAL SHAPE INDEX, will be within the conservative envelope of operating conditions consistent with the safety analysis assumptions and which have been analytically demonstrated adequate to maintain an acceptable minimum DNBR throughout all anticipated operational occurrences. Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained in the event of a loss of flow transient.

The purpose of T.S. 3.3.1 is to ensure that (1) the associated Engineered Safety Features Actuation action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

C. NEED FOR THE TECHNICAL SPECIFICATION AMENDMENT

The proposed changes are due to (1) ensuring operation of the reactor within approved safety analysis for Cycle 2 by modifying the T.S. graphs, (2) increasing operator reliability by placing DNBR operating limits in one place, and (3) eliminating superfluous information to reduce confusion and the possibility of misuse. (i.e., eliminating the Table in T.S. 4.2.4.4)

D. BASIS FOR PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

1. The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety.

   A discussion of these standards as they relate to the amendment request follows:

   Standard 1--Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change to the graphs of T.S. 3.2.4 does not involve a significant increase in the probability or consequences of an accident previously evaluated because the Cycle 2 safety analyses have shown that when COLSS is in service and at least one CEAC is operable, Specification 3.2.4a provides enough margin to DNB to accommodate the limiting Anticipated Operational Occurrence (AOO) without violating the Specified Acceptable Fuel Design Limits (SAFDL). For the case when neither CEAC is operable but COLSS is in service, the CPCs assume a preset CEA configuration because they can not obtain the required CEA position information to ensure that the SAFDL or DNBR will not be violated during an AOO. Thus, as a result of the reevaluation of the limiting AOOs for Cycle 2, Specification 3.2.4.b requires that core power be reduced to a value, (based on Figure 3.2-1) less than the current COLSS calculated power operating limit. This ensures the limiting AOO will not result in a violation of SAFDLs. The proposed revision to Figure 3.2-2 accounts for the situation when COLSS is out-of-service but at least one CEAC is operable. In this case, the Cycle 2 safety analysis has shown that, by maintaining the CPC calculated DNBR above the value shown in the figure, the limiting AOO will not result in a violation of the SAFDLs. When COLSS is out of service and both CEACs are inoperable, there must be additional margin to DNB set aside in the CPCs to ensure they can mitigate the consequences of the limiting AOO. A reevaluation of the limiting transients performed as part of the Cycle 2 safety analysis has shown that, by maintaining the CPC calculated DNBR above the limits shown in the proposed Figure 3.2-2a, there is sufficient thermal margin to ensure that the limiting AOO will not result in a violation of the SAFDLs. Therefore, the proposed change will not significantly increase the probability or consequences of any accident previously evaluated.

The proposed change to the format of T.S. 3.2.4 and 3.3.1 does not involve a significant increase in the probability or consequences of an accident previously evaluated because consolidation of the DNBR operating limits within one Technical Specification will increase the operator's ability to ensure proper operation of the reactor. The proposed format change still contains the same Limiting Conditions for Operations (LCO), ACTIONS and surveillance requirements as the original Technical Specifications. Therefore, the change will not significantly increase the probability or consequences of any accident previously evaluated.

The proposed change to eliminate the DNBR penalty factors table of T.S. 4.2.4.4 does not involve a significant increase in the probability or consequences of an accident previously evaluated because the penalty is an allowance for rod bow and has been incorporated into the DNBR value for Cycle 2. This can be done because the burnup of the reactor core in Cycle 2 will reach the value for applying the maximum rod bow penalty and the table will no longer be needed (see Attachment 12). Therefore, the change will not significantly increase the probability or consequences of any accident previously evaluated.

Standard 2--Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change to the graphs of T.S. 3.2.4 will not create the possibility of a new or different kind of accident from any accident previously evaluated because operation of the reactor within the limits as set forth in the graphs ensures that the reactor will not exceed the SAFDLs as defined for the reference cycle (Cycle 1) during Cycle 2. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated will not be created.

The proposed change to the format of T.S. 3.2.4 and 3.3.1 will not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed change reduces the possibility of human error by consolidating closely related allowable operations into a single entity and by clearly identifying each allowable operation. The contents of the proposed T.S. are the same as those of T.S. 3.2.4 and 3.3.1, thus, the only change is in regard to the human factors element. Therefore, by keeping the same contents but arranging them so as to reduce human error, the proposed change will not create the possibility of a new or different kind of accident not previously evaluated.

The proposed change to eliminate the DNBR penalty factors table of T.S. 4.2.4.4 will not create the possibility of a new or different kind of accident from any accident previously evaluated because the possibility of misusing the table is eliminated.

Standard 3--Involve a significant reduction in a margin of safety.

The proposed change to the graphs of T.S. 3.2.4 does not involve a significant reduction in a margin of safety because the change is to ensure that there will always be sufficient margin to DNBR such that the CPCs can mitigate the consequences of violating the SAFDLs. Figures 3.2-1, 3.3-2, and 3.2-2a represent a conservative envelope of operating conditions for the CPCs and COLSS which is consistent with Cycle 2 safety analysis assumptions. This band of operating conditions has been analytically demonstrated to maintain an acceptable minimum DNBR throughout all AOOs. Therefore, the proposed change does not reduce the margin of safety.

The proposed change to the format of T.S. 3.2.4 and 3.3.1 does not involve a significnat reduction in a margin of safety because the contents of the Technical Specifications have remained the same, only a rearrangment of information has taken place. Therefore, the proposed change does not reduce the margin of safety.

The proposed change to eliminate the DNBR penalty factors table of T.S. 4.2.4.4 does not involve a significant reduction in a margin of safety because the maximum rod bow penalty factor has been applied to the DNBR value for Cycle 2 and, therefore, the table is no longer needed and the margin of safety has been maintained for Cycle 2.

2. The proposed amendment matches the guidance concerning the application of standards for determining whether or not a significant hazards consideration exists (51 FR 7751) by examples:

(i) A purely administrative change to Technical Specification: for example, a change to achieve consistency throughout the Technical Specifications, in correction of an error, or a change in nomenclature.

and

(iii) For a nuclear power reactor, a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptable criteria for the Technical Specifications, the analytical methods used to demonstrate conformance with the Technical Specifications and regulations are not significantly changed, and that NRC has previously found such methods acceptable.

E. SAFETY EVALUATION FOR THE AMENDMENT REQUEST

The proposed Technical Specification amendment will not increase the probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the FSAR. The proposed change to the graphs of T.S. 3.2.4 ensures that the reactor will be operated within a conservative envelope of operating conditions, consistent with the safety analysis, during Cycle 2, thus ensuring no increase in the probability of occurrence or the consequences of an accident or malfunction.

The changes to the format of T.S. 3.2.4 will increase the operator's ability to ensure correct operation of the reactor by consolidating related operation requirements into one Technical Specification. Because the change does not change the LCO, ACTIONS or surveillance requirements only the manner of presentation, no increase in the probability of occurrence or the consequences of an accident or malfunction will be experienced. The proposed change to eliminate the DNBR rod bow penalty factors table of T.S. 4.2.4.4 reduces confusion since the table is no longer needed. Because the maximum rod bow penalty factor has been incorporated into the Cycle 2 DNBR value no increase in the probability of occurrence or the consequences of an accident or malfunction will be incurred when the table has been deleted.

The proposed Technical Specification amendment will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR. The proposed changes to the graphs of T.S. 3.2.4 ensure the operation of the reactor, during Cycle 2 operation, to be within the same limits as for Cycle 1. Therefore, the possibility for an accident or malfunction of a different type will not be created. The proposed changes to the format of T.S. 3.2.4 do not change the LCOs, ACTIONS or surveillance requirements of the T.S., only the manner of presentation, thus the change does not create the possibility of an accident or malfunction of a different kind to occur. The proposed change to eliminate the rod bow penalty factors of T.S. 4.2.4.4. removes information no longer needed or necessary. A maximum rod bow penalty has been applied to the DNBR value, therefore, the change will not create the possibility for an accident or malfunction of a different kind to occur.

The proposed Technical Specification amendment will not reduce the margin of safety as defined in the basis for the Technical Specifications. The proposed changes either ensure sufficient margin will be maintained or do not change LCOs, actions or surveillance requirements required to maintain the margin of safety. Therefore, the margin of safety is not reduced.

F. ENVIRONMENTAL IMPACT CONSIDERATION DETERMINATION

The proposed change request does not involve an unreviewed environmental question because operation of PVNGS Unit 2, in accordance with this change, would not:

1. Result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the staff's testimony to the Atomic Safety and Licensing Board; or

2. Result in a significant change in effluents or power levels; or

3. Result in matters not previously reviewed in the licensing basis for PVNGS which may have a significant environmental impact.

G. MARKED-UP TECHNICAL SPECIFICATION CHANGE PAGES

Limiting Conditions For Operation And Surveillance Requirements:

XIX
3/4 2-5
3/4 2-6
3/4 2-7
3/4 2-7a
3/4 3-7
3/4 3-8
3/4 3-9
3/4 3-10
B 3/4 2-3
B 3/4 1-6

LIST OF FIGURES

add

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by operating within the Region of Acceptable Operation of Figure 3.2-1 or 3.2-2, as applicable, or in accordance with the requirements of Action 6 of Table 3.3-1.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel below the DNBR limit, within 15 minutes initiate corrective action to restore either the DNBR core power operating limit or the DNBR to within the limits and either:

a. Restore the DNBR core power operating limit or DNBR to within its limits within 1 hour, or

b. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR margin, as indicated on all OPERABLE DNBR margin channels, is within the limit shown on Figure 3.2-2.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

4.2.4.4 The following DNBR or equivalent penalty factors shall be verified to be included in the COLSS and CPC DNBR calculations at least once per 31 EFPD.

Burnup $\left(\frac{GWD}{MTU}\right)$	DNBR Penalty (%)*
0-10	0.5
10-20	1.0
20-30	2.0
30-40	3.5
40-50	5.5

*The penalty for each batch will be determined from the batch's maximum burnup assembly and applied to the batch's maximum radial power peak assembly. A single net penalty for COLSS and CPC will be determined from the penalties associated with each batch accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

POWER DISTRIBUTION LIMITS

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by one of the following methods:

a. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR (when COLSS is in service, and either one or both CEACs are operable); or

b. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR decreased by the allowance shown in Figure 3.2-1 ~~10% RATED THERMAL POWER~~ (when COLSS is in service and neither CEAC is operable); or

c. Operating within the region of acceptable operation of Figure 3.2-2 using any operable CPC channel (when COLSS is out of service and either one or both CEACs are operable); or

d. Operating within the region of acceptable operation of Figure 3.2-~~3~~2a using any operable CPC channel (when COLSS is out of service and neither CEAC is operable).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With the DNBR not being maintained:

1. As indicated by COLSS calculated core power exceeding the appropriate COLSS calculated power operating limit; or

2. With COLSS out of service, operation outside the region of acceptable operation of Figure 3.2-2 or 3.2-~~3~~2a, as applicable;

within 15 minutes inititate corrective action to increase the DNBR to within the limits and either:

a. Restore the DNBR to within its limits within 1 hour, or

b. ~~Be in at least HOT STANDBY within the next 6 hours.~~ Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR, as indicated on any OPERABLE DNBR channel, is within the limit shown on Figure 3.2-2 or Figure 3.2-~~3~~2a.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

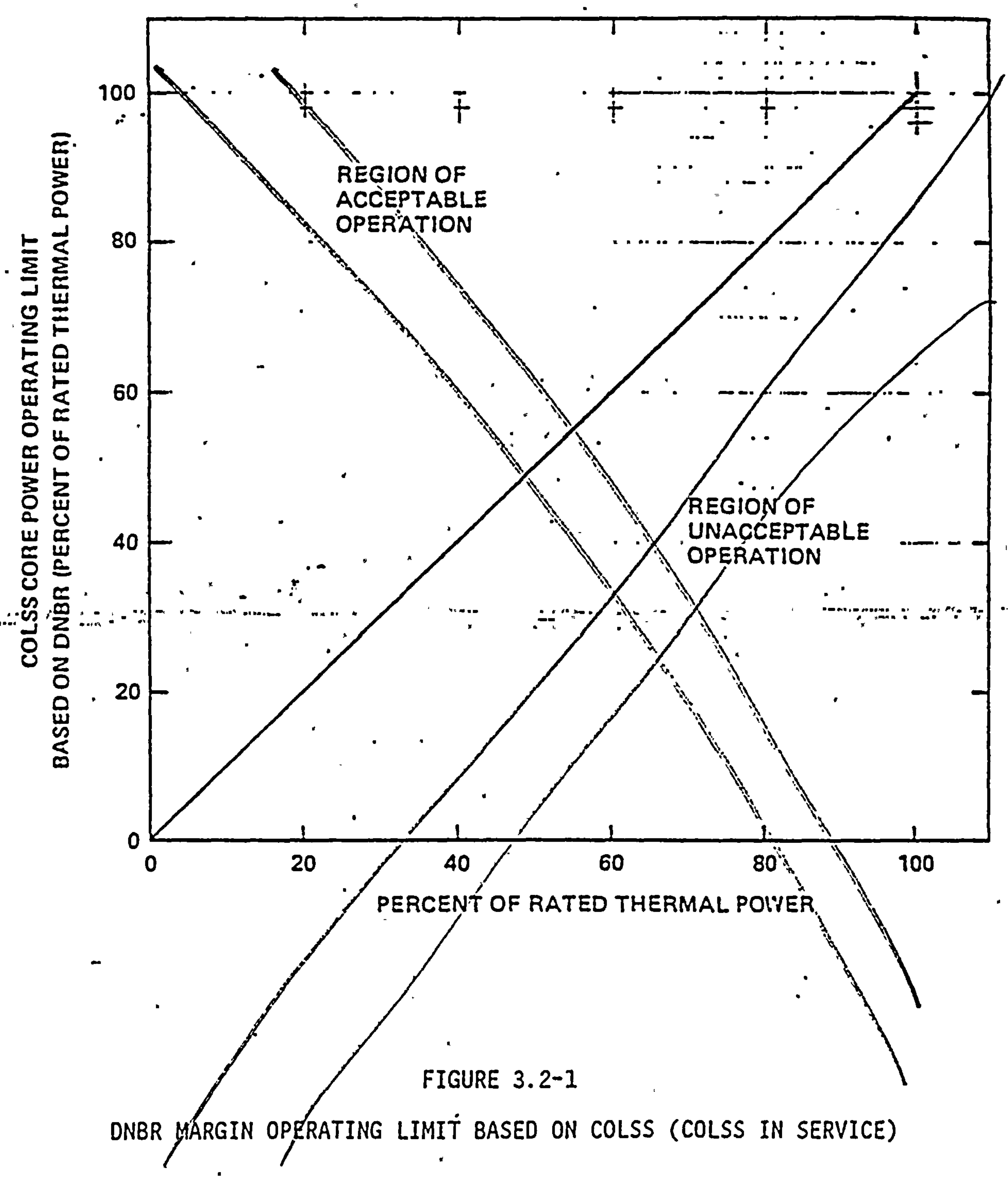


FIGURE 3.2-1

DNBR MARGIN OPERATING LIMIT BASED ON COLSS (COLSS IN SERVICE)

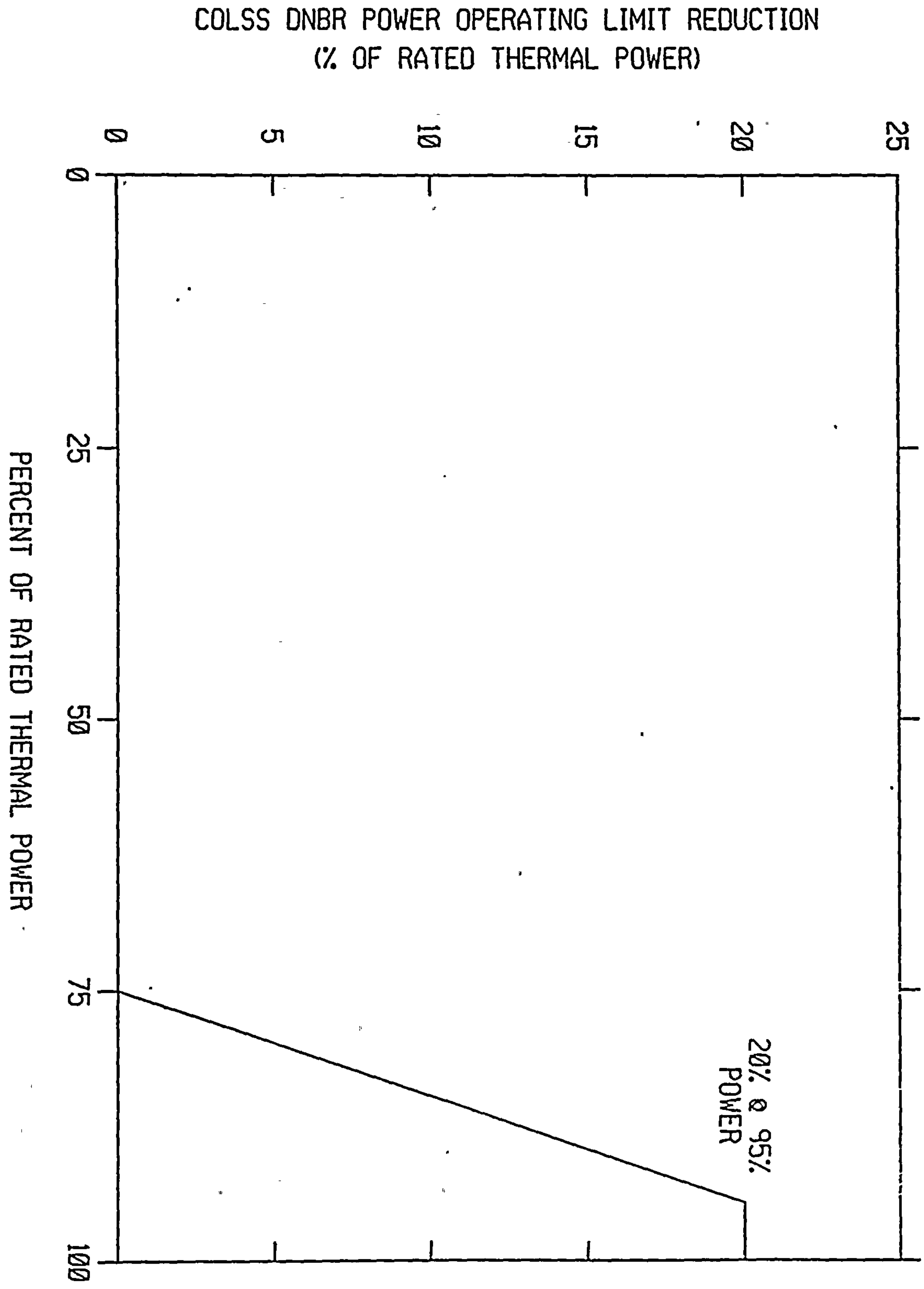
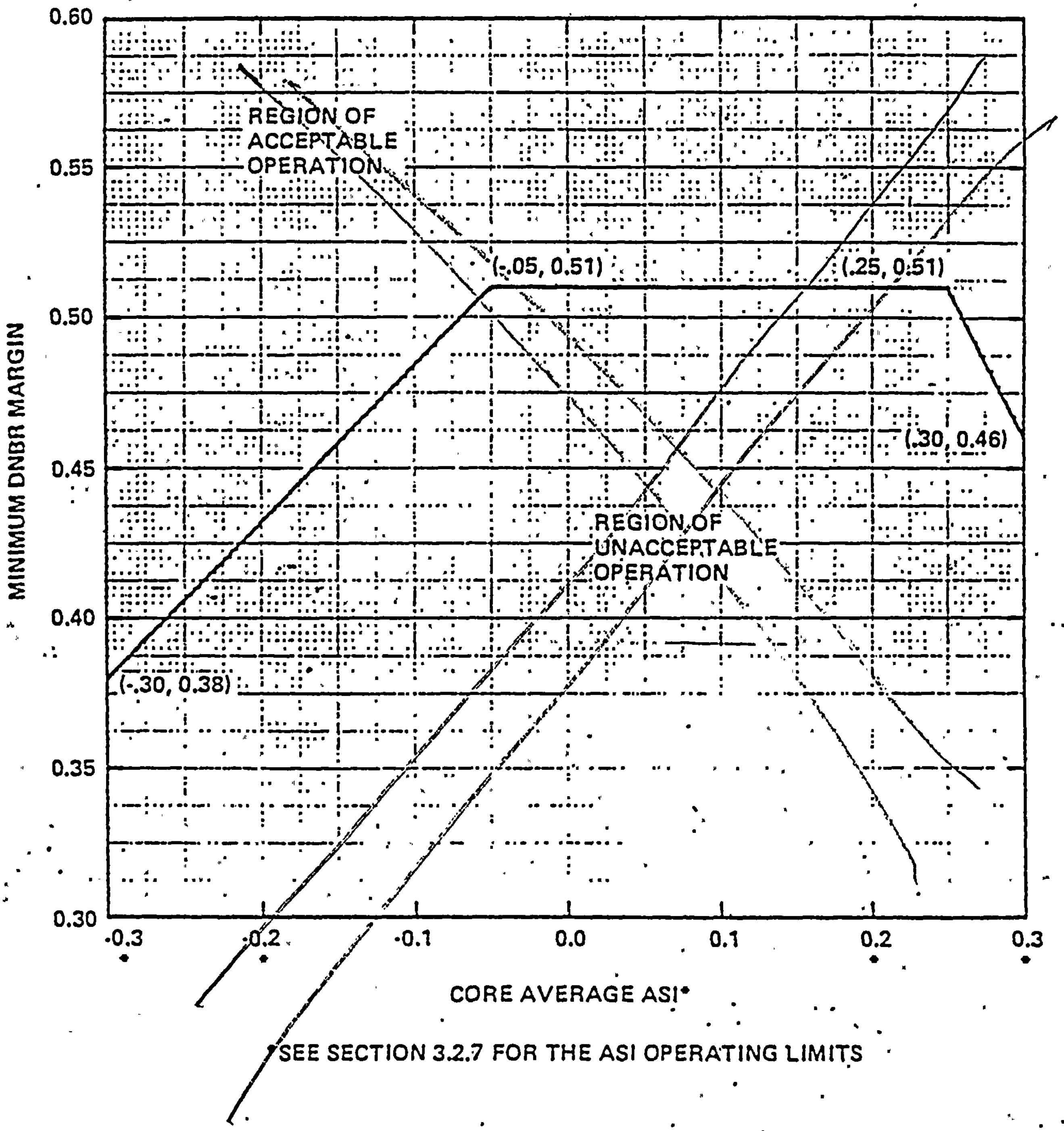


FIGURE 3.2-1

COLSS DNBR POWER OPERATING LIMIT
ALLOWANCE FOR BOTH CEAC'S INOPERABLE



*SEE SECTION 3.2.7 FOR THE ASI OPERATING LIMITS

FIGURE 3.2-2

DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS

(COLSS OUT OF SERVICE)

CONTROLLED BY USER

COLSS OUT OF SERVICE DNBR LIMIT LINE

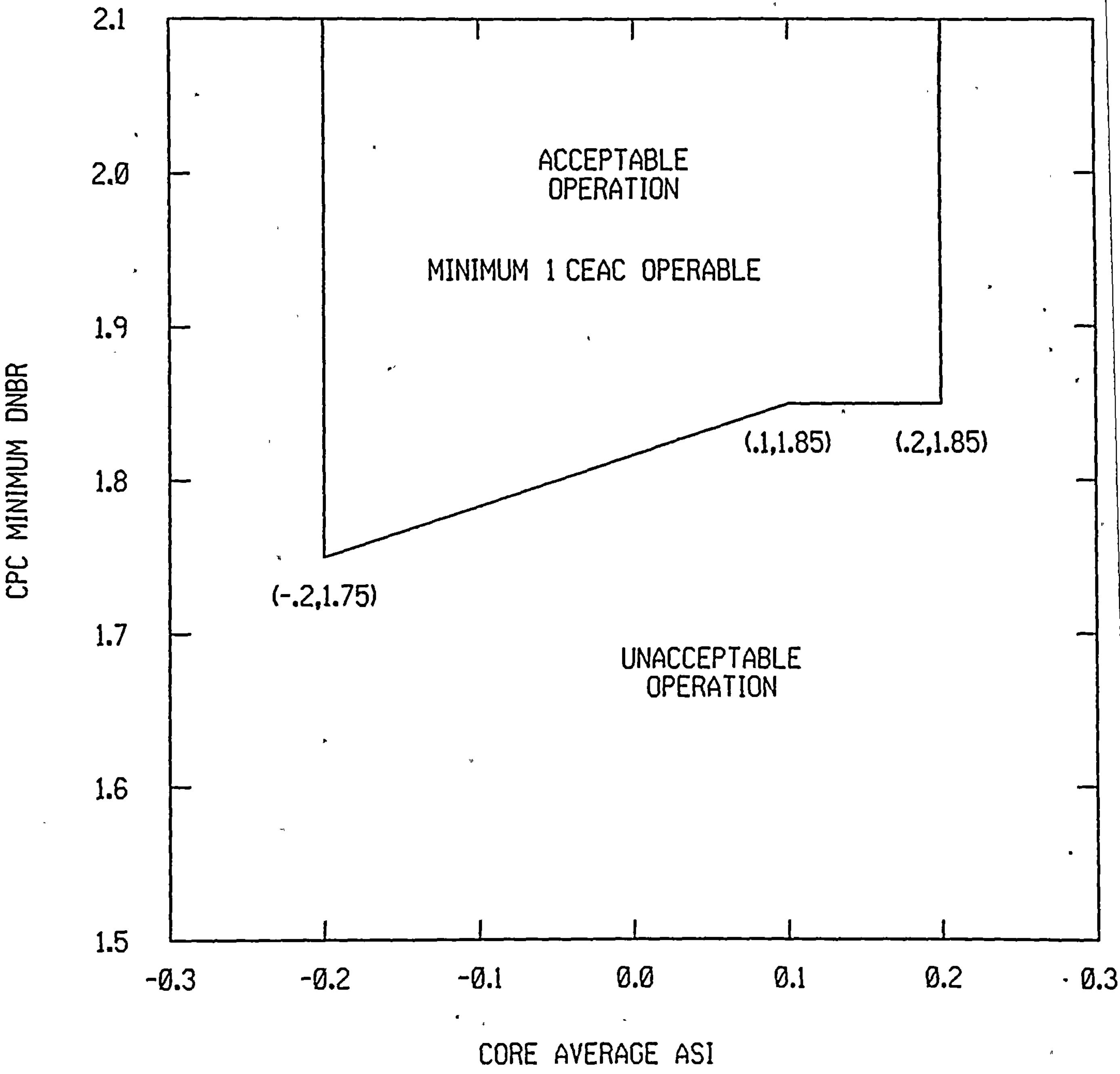


FIGURE 3.2-2

DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS (COLSS OUT OF SERVICE, CEAC'S OPERABLE)

COLSS OUT OF SERVICE DNBR LIMIT LINE

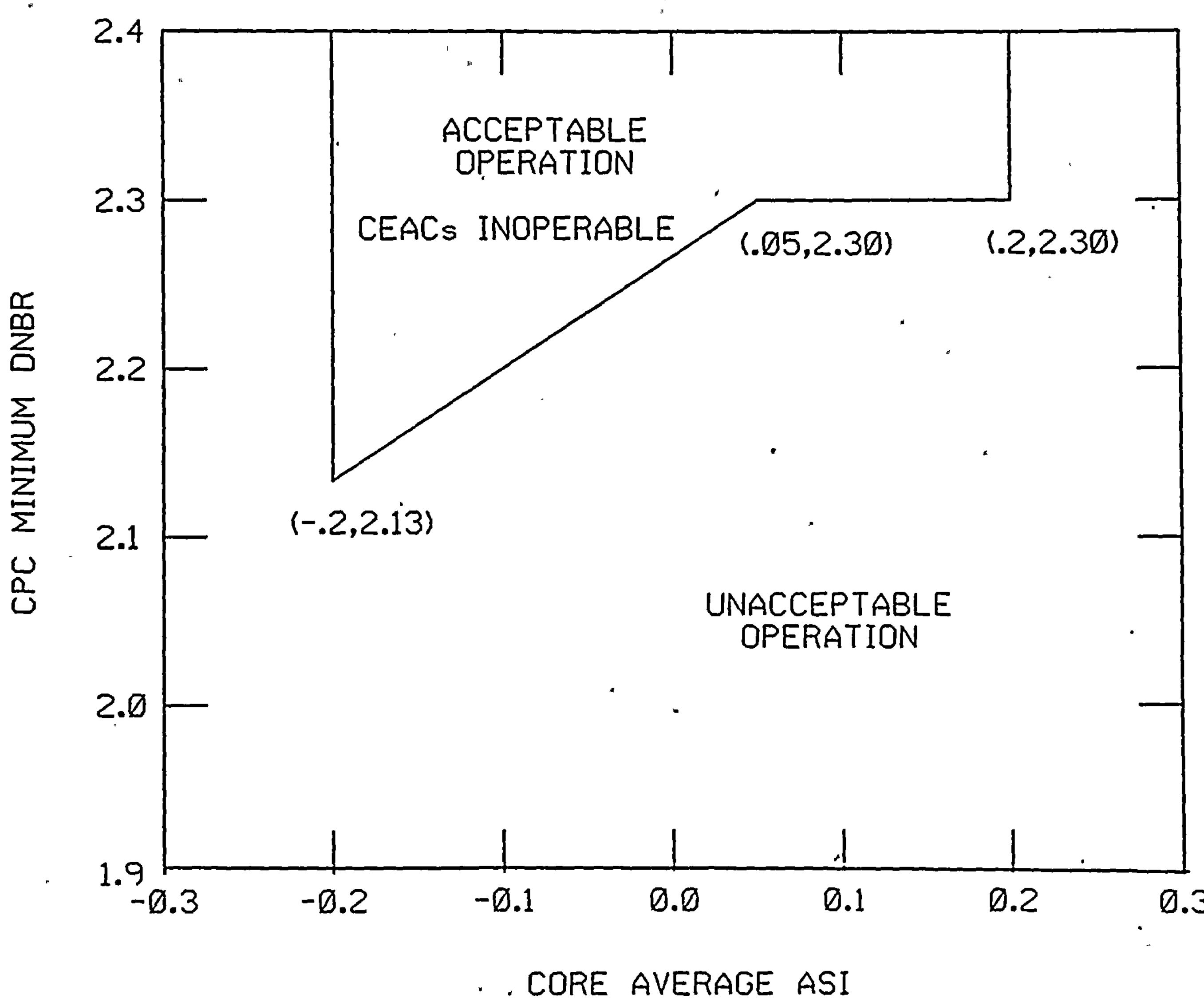


FIGURE 3.2-2a

DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS
(COLSS OUT OF SERVICE,CEACs INOPERABLE)

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

ACTION STATEMENTS

3.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
4.	Steam Generator Level - Low (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
5.	Core Protection Calculator	Local Power Density - High (RPS) DNBR - Low (RPS)

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the reactor trip breaker of the inoperable channel is placed in the tripped condition within 1 hour, otherwise, be in at least HOT STANDBY within 6 hours; however, the trip breaker associated with the inoperable channel may be closed for up to 1 hour for surveillance testing per Specification 4.3.1.1.

ACTION 6 - a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 6.6 inches (indicated position) of all other CEAs in its group. After 7 days, operation may continue provided that the conditions of Action Item 6.b ~~or 6.c~~ are met.

b. With both CEACs inoperable ~~and COLSS in service~~, operation may continue provided that:

~~1. Within 1 hour:~~

~~a) Operation is restricted to the limits shown in Figure 3.3-1. The DNBR margin required by Specification 3.2.4 is replaced by this restriction when both CEAC's are inoperable and COLSS is in operation.~~

~~b) The Linear Heat Rate Margin required by Specification 3.2.1 is maintained.~~

~~c) The Reactor Power Cutback System is placed out of service.~~

CONTROLLED BY USER

ACTION STATEMENTS

add → 1. Within 1 hour the DNBR margin required by Specification 3.2.4b (COLSS in service) or 3.2.4d (COLSS out of service) is satisfied and the Reactor Power Cutback System is disabled, and

2. Within 4 hours:

   a) All full-length and part-length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 5 may be inserted no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to be indicated that both CEAC's are inoperable.

   c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Standby" mode except during CEA group 5 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full-length and part-length CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 5 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 6.6 inches (indicated position) of all other CEAs in its group.

DELETED

~~4. Following a CEA misalignment with both CEAC's inoperable and COLSS in operation, operation may continue provided that within 1 hour:~~

~~The power is reduced to 85% of the pre-misaligned power but need not be reduced to less than 50% of RATED THERMAL POWER. This power restriction replaces the power restriction of Specification 3.1.3.1, Figure 3.1-2B, otherwise Specification 3.1.3.1 remains applicable.~~

c. ~~With both CEACs inoperable and COLSS out-of-service, operation may continue provided that:~~

~~1. Within 1 hour:~~

   ~~a) The existing CPC value of the CPC addressable constant "BERR1" is multipled by 1.19 and the resulting value is re-entered into the CPCs.~~

   ~~b) The Reactor Power Cutback System is placed out of service~~

   ~~c) The COLSS out of service Limit Line, on Figure 3.2-2 of Specification 3.2.4, is not applicable to this mode of operation.~~

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

ACTION STATEMENTS

Deleted

2. Within 4 hours:

   a) All full length and part length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 5 may be inserted no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to be indicated that both CEAC's are inoperable.

   c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Standby" mode except during CEA group 5 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full length and part length CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 5 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 6.6 inches (indicated position) of all other CEAs in its group.

4. Following a CEA misalignment with both CEAC's and COLSS inoperable, operation may continue provided that within 1 hour:

   The power is reduced to 85% of the pre-misaligned power but need not be reduced to less than 50% of RATED THERMAL POWER. This power restriction replaces the power restriction of Specification 3.1.3.1, Figure 3.1-2B, otherwise Specification 3.1.3.1 remains applicable.

Deleted

ACTION 7 - With three or more auto restarts, excluding periodic auto restarts (Code 30 and Code 33), of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore an inoperable channel to OPERABLE status within 48 hours or open an affected reactor trip breaker within the next hour.

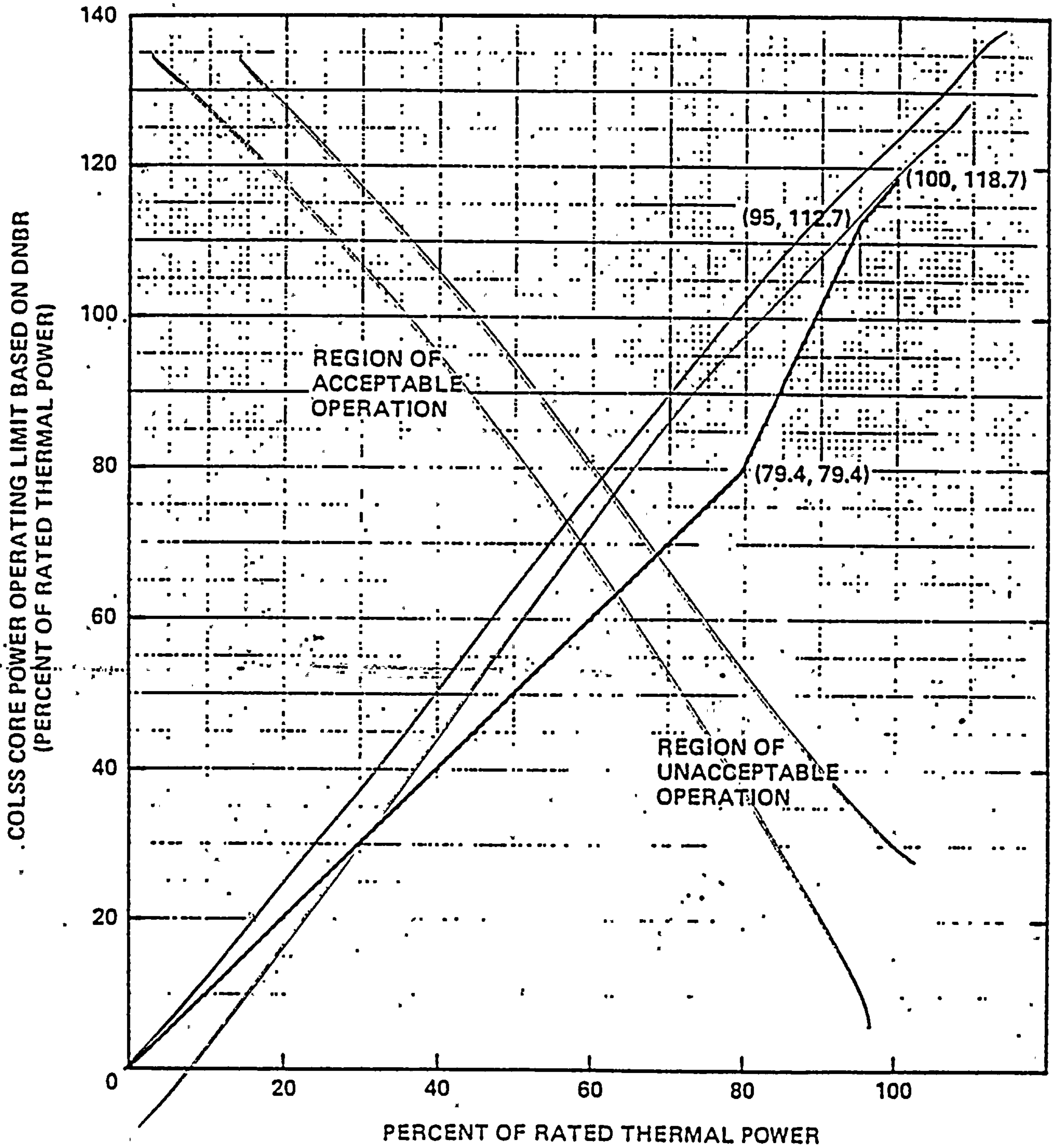


FIGURE 3.3-1

DNBR MARGIN OPERATING LIMIT BASED ON COLSS

FOR BOTH CEACs INOPERABLE

AZIMUTHAL POWER TILT - T_q (Continued)

P_{tilt}/P_{untilt} is the ratio of the power at a core location in the presence of a tilt to the power at that location with no tilt.

The AZIMUTHAL POWER TILT allowance used in the CPCs is defined as the value of CPC addressable constant TR-1.0.

3/4.2.4 DNBR MARGIN

The limitation on DNBR as a function of AXIAL SHAPE INDEX represents a conservative envelope of operating conditions consistent with the safety analysis assumptions and which have been analytically demonstrated adequate to maintain an acceptable minimum DNBR throughout all anticipated operational occurrences, ~~of which the loss of flow transient is the most limiting~~. Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained in the event of a loss of flow transient.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the DNBR channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the DNBR does not violate its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core operating limit corresponding to the allowable minimum DNBR. ~~Reactor operation at or below this calculated power level assures that the limits of Figure 3.2-1 are not violated.~~ The COLSS calculation of core power operating limit based on DNBR includes appropriate penalty factors which provide, with a 95/95 probability/confidence level, that the core power limits calculated by COLSS (based on the minimum DNBR Limit) is conservative with respect to the actual core power limit. These penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering heat flux, state parameter measurement, software algorithm modelling, computer processing, rod bow, and core power measurement.

Parameters required to maintain the margin to DNB and total core power are also monitored by the CPCs. Therefore, in the event that the COLSS is not being used, operation within the limits of Figures ~~3.2-2~~ 3.2.2 and 3.2.2a can be maintained by utilizing a predetermined DNBR as a function of AXIAL SHAPE INDEX and by monitoring the CPC trip channels. The above listed uncertainty and penalty factors are also included in the CPCs which assume a minimum core power of 20% of RATED THERMAL POWER. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being less ~~in~~accurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings.

A ~~The~~ DNBR penalty factor~~s~~ ~~listed in Specification 4.2.4.4 are penalties used~~ has been included in the COLSS and CPC DNBR calculations to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher average burnup will experience a greater magnitude of rod bow. Conversely, lower burnup assemblies will experience less rod bow. In design calculations, the penalty for each batch required to compensate for rod bow is determined from a batch's maximum average assembly burnup applied to the batch's maximum integrated planar-radial power peak. A single net penalty for COLSS and CPC is then determined from the penalties associated with each batch, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

MOVABLE CONTROL ASSEMBLIES (Continued)

and load maneuvering. Analyses are performed based on the expected mode of operation of the NSSS (base load maneuvering, etc.) and from these analyses CEA insertions are determined and a consistent set of radial peaking factors defined. The Long Term Steady State and Short Term Insertion Limits are determined based upon the assumed mode of operation used in the analyses and provide a means of preserving the assumptions on CEA insertions used. The limits specified serve to limit the behavior of the radial peaking factors within the bounds determined from analysis. The actions specified serve to limit the extent of radial xenon redistribution effects to those accommodated in the analyses. The Long and Short Term Insertion Limits of Specification 3.1.3.6 are specified for the plant which has been designed for primarily base loaded operation but which has the ability to accommodate a limited amount of load maneuvering.

The Transient Insertion Limits of Specification 3.1.3.6 and the Shutdown CEA Insertion Limits of Specification 3.1.3.5 ensure that (1) the minimum SHUTDOWN MARGIN is maintained, and (2) the potential effects of a CEA ejection accident are limited to acceptable levels. Long-term operation at the Transient Insertion Limits is not permitted since such operation could have effects on the core power distribution which could invalidate assumptions used to determine the behavior of the radial peaking factors.

The PVNGS CPC and COLSS systems are responsible for the safety and monitoring functions, respectively, of the reactor core. COLSS monitors the DNB Power Operating Limit (POL) and various operating parameters to help the operator maintain plant operation within the limiting conditions for operation (LCO). Operating within the LCO guarantees that in the event of an Anticipated Operational Occurrence (AOO), the CPCs will provide a reactor trip in time to prevent unacceptable fuel damage.

and CEA misoperation

The COLSS reserves the Required Overpower Margin (ROPM) to account for the Loss of Flow (LOF) transients ~~which is the limiting AOO for the PVNGS plants.~~ When the COLSS is Out of Service (COOS), the monitoring function is performed via the CPC calculation of DNBR in conjunction with ~~a~~ Technical Specification COOS Limit Lines (Figures 3.2-2) which restricts the reactor power sufficiently to preserve the ROPM. and 3.2-2a

The reduction of the CEA deviation penalties in accordance with the CEAC (Control Element Assembly Calculator) sensitivity reduction program has been performed. This task involved setting many of the inward single CEA deviation penalty factors to 1.0. An inward CEA deviation event in effect would not be accompanied by the application of the CEA deviation penalty in either the CPC DNB and LHR (Linear Heat Rate) calculations for those CEAs with the reduced penalty factors. The protection for an inward CEA deviation event is thus accounted for separately.

ATTACHMENT 12

A. DESCRIPTION OF THE TECHNICAL SPECIFICATION AMENDMENT REQUEST

The proposed amendment change expands the operating limits of Azimuthal Tilt with COLSS in service. The azimuthal tilt limits will be a step function of power with the upper limit of 0.20 at 20% power and stepping down to 0.10 at 40% power, where it remains steady through to 100% power.

B. PURPOSE OF THE TECHNICAL SPECIFICATION

The limitations on the Azimuthal Power Tilt are to ensure that design safety margins are maintained.

C. NEED FOR THE TECHNICAL SPECIFICATION AMENDMENT

During a reactor power cutback event in Unit 1 the plant was unable to go above 20% power because the azimuthal tilt limit would have been exceeded. They were required to remain below 20% power for approximately 5 hours until xenon burned out. This delay could have been prevented and the azimuthal tilt corrected if the plant had been allowed to increase power. This would cause the xenon to burn out faster thus restoring the plant within the limits sooner. By implementing the proposed change such delays could be avoided.

D. BASIS FOR PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

1. The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety.

   A discussion of these standards as they relate to the amendment request follows:

   Standard 1--Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated because a reevaluation of the safety analysis pertaining to azimuthal tilt was conducted and the results of the reanalysis show that for the conditions of azimuthal tilt as defined in the new Figure 3.2-1A the safety analysis of the referenced cycle (Cycle 1) is bounding. Therefore there is no change to the probability or consequences of an accident previously evaluated in the FSAR.

Standard 2--Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated. The results of the reanalysis were found to be bounded by the reference cycle safety analysis. Relaxing the azimuthal power tilt limit at lower power levels will not create any new or different kinds of accidents.

Standard 3--Involve a significant reduction in a margin of safety.

The proposed change does not involve a significant reduction in the margin of safety. A reanalysis was performed using the proposed tilt limits and it was found that the results of the reanalysis were bounded by the reference cycle safety analysis. Therefore the margin of safety is maintained.

2. The proposed amendment matches the guidance concerning the application of standards for determining whether or not a significant hazards consideration exists (51 FR 7751) by example:

   (vi) A change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan: for example, a change resulting from the application of a small refinement of a previously used calculation model or design method.

E. SAFETY EVALUATION FOR THE AMENDMENT REQUEST

The proposed Technical Specification amendment will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR. The proposed change does not change or replace equipment or components important to safety. The change is bounded by the existing safety analysis and will not increase the probability of an occurrence or consequences of an accident.

The proposed Technical Specification amendment will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR. By determining that the results of the reanalysis were bounded by the reference cycle safety analysis the field of accidents or malfunctions have not changed. Therefore there is no increase in the probability for an accident or malfunction of a different type than any previously evaluated in the FSAR.

The proposed Technical Specification amendment will not reduce the margin of safety as defined in the basis for the Technical Specifications. To determine the impact of the change to the azimuthal tilt limits, a reanalysis was performed. The results of the reanalysis were bounded by the reference cycle safety analysis and therefore the margin of safety has been maintained.

F. ENVIRONMENTAL IMPACT CONSIDERATION DETERMINATION

The proposed change request does not involve an unreviewed environmental question because operation of PVNGS Unit 2, in accordance with this change, would not:

1. Result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the staff's testimony to the Atomic Safety and Licensing Board; or

2. Result in a significant change in effluents or power levels; or

3. Result in matters not previously reviewed in the licensing basis for PVNGS which may have a significant environmental impact.

G. MARKED-UP TECHNICAL SPECIFICATION CHANGE PAGES

Limiting Conditions For Operation And Surveillance Requirements:

3/4 2-3 B 3/4 2-2
3/4 2-4

LIST OF FIGURES

3/4.2.3 AZIMUTHAL POWER TILT - T_q

LIMITING CONDITION FOR OPERATION

3.2.3 The AZIMUTHAL POWER TILT (T_q) shall be less than or equal to the AZIMUTHAL POWER TILT Allowance used in the Core Protection Calculators (CPCs).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER*.

ACTION:

a. With the measured AZIMUTHAL POWER TILT determined to exceed the AZIMUTHAL POWER TILT Allowance used in the CPCs but less than or equal to ~~0.10~~, within 2 hours either correct the power tilt or adjust the AZIMUTHAL POWER TILT Allowance used in the CPCs to greater than or equal to the measured value.

The limit in Figure 3.2-1A with COLSS in service or 0.10 with COLSS out of Service

b. With the measured AZIMUTHAL POWER TILT determined to exceed ~~0.10~~:

1. Due to misalignment of either a part-length or full-length CEA, within 30 minutes verify that the Core Operating Limit Supervisory System (COLSS) (when COLSS is being used to monitor the core power distribution per Specifications 4.2.1 and 4.2.4) is detecting the CEA misalignment.

2. Verify that the AZIMUTHAL POWER TILT is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and verify that the Variable Overpower Trip Setpoint has been reduced as appropriate within the next 4 hours.

3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the AZIMUTHAL POWER TILT is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

*See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The AZIMUTHAL POWER TILT shall be determined to be within the limit above 20% of RATED THERMAL POWER by:

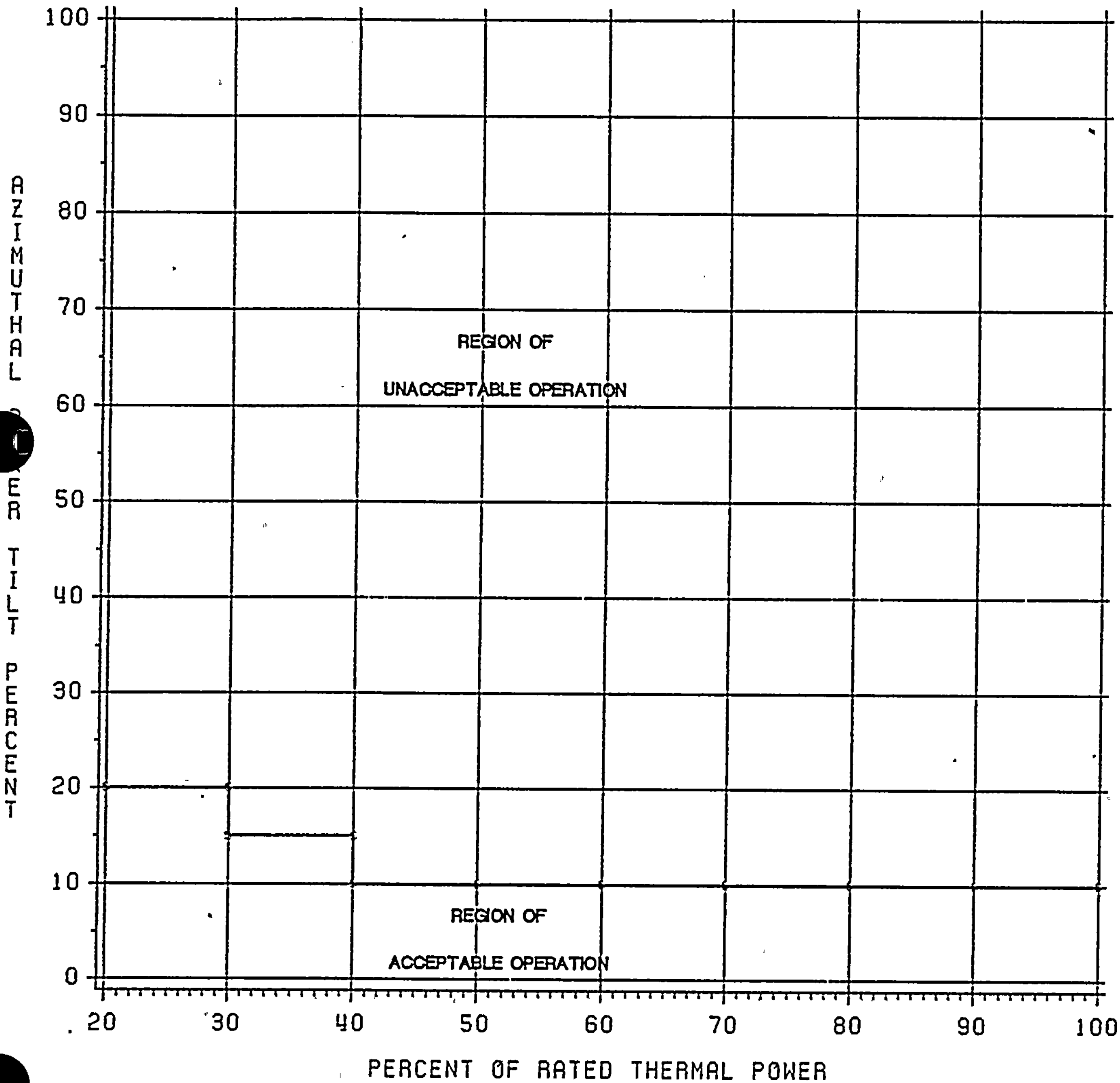
a. Continuously monitoring the tilt with COLSS when the COLSS is ~~OPERABLE~~ in service.

b. Calculating the tilt at least once per 12 hours when the COLSS is ~~inoperable~~. out of service.

c. Verifying at least once per 31 days, that the COLSS Azimuthal Tilt Alarm is actuated at an AZIMUTHAL POWER TILT less than or equal to the AZIMUTHAL POWER TILT Allowance used in the CPCs.

d. Using the incore detectors at least once per 31 EFPD to independently confirm the validity of the COLSS calculated AZIMUTHAL POWER TILT.

FIGURE 3.2 – 1A
AZIMUTHAL POWER TILT LIMIT
VS
THERMAL POWER
(COLSS IN SERVICE)



PALO VERDE - UNIT 2

3/4.2.2 PLANAR RADIAL PEAKING FACTORS

Limiting the values of the PLANAR RADIAL PEAKING FACTORS (F_{xy}^{c}) used in the COLSS and CPCs to values equal to or greater than the measured PLANAR RADIAL PEAKING FACTORS (F_{xy}^{m}) provides assurance that the limits calculated by COLSS and the CPCs remain valid. Data from the incore detectors are used for determining the measured PLANAR RADIAL PEAKING FACTORS. A minimum core power at 20% of RATED THERMAL POWER is assumed in determining the PLANAR RADIAL PEAKING FACTORS. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The periodic surveillance requirements for determining the measured PLANAR RADIAL PEAKING FACTORS provides assurance that the PLANAR RADIAL PEAKING FACTORS used in COLSS and the CPCs remain valid throughout the fuel cycle. Determining the measured PLANAR RADIAL PEAKING FACTORS after each fuel loading prior to exceeding 70% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

3/4.2.3 AZIMUTHAL POWER TILT - T_q

The limit Figure 3.2-1A with COLSS in service or 0.10 with COLSS out of Service

The limitations on the AZIMUTHAL POWER TILT are provided to ensure that design safety margins are maintained. An AZIMUTHAL POWER TILT greater than ~~0.10~~ is not expected and if it should occur, operation is restricted to only those conditions required to identify the cause of the tilt. The tilt is normally calculated by COLSS. A minimum core power of 20% of RATED THERMAL POWER is assumed by the CPCs in its input to COLSS for calculation of AZIMUTHAL POWER TILT. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The surveillance requirements specified when COLSS is out of service provide an acceptable means of detecting the presence of a steady-state tilt. It is necessary to explicitly account for power asymmetries because the radial peaking factors used in the core power distribution calculations are based on an untilted power distribution.

The AZIMUTHAL POWER TILT is equal to $(P_{tilt}/P_{untilt})-1.0$ where:

AZIMUTHAL POWER TILT is measured by assuming that the ratio of the power at any core location in the presence of a tilt to the untilted power at the location is of the form:

$$P_{tilt}/P_{untilt} = 1 + T_q \, g \cos(\Theta - \Theta_0)$$

where:

T_q is the peak fractional tilt amplitude at the core periphery

g is the radial normalizing factor

Θ is the azimuthal core location

Θ_0 is the azimuthal core location of maximum tilt

ATTACHMENT 13

A. DESCRIPTION OF THE TECHNICAL SPECIFICATION AMENDMENT REQUEST

The proposed amendment ensures the Refueling Actuation Signal (RAS) trip value of the Refueling Water Storage Tank for recirculation is maintained at the midpoint of the allowable operational values by removing the "greater than" sign from the trip value as set forth in Technical Specification (T.S.) 3.3.2 Table 3.3-4.

B. PURPOSE OF THE TECHNICAL SPECIFICATION

The purpose of T.S. 3.3.2 is to ensure that (1) the associated Engineered Safety Features Actuation action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

C. NEED FOR THE TECHNICAL SPECIFICATION AMENDMENT

The proposed change to T.S. 3.3.2 Table 3.3-4 will eliminate an abiquity concerning the level setpoint in relation to the allowable range.

D. BASIS FOR PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

1. The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety.

   A discussion of these standards, as they relate to the amendment request, follows:

   Standard 1--Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated because, by maintaining the RAS trip value at the midpoint of the allowable band, the proposed change is more restrictive. This, in turn, limits the

operation of the Refueling Water Storage Tank such that a maximum assurance of protecting the pumps from cavitating is provided. Since the change is still within the limits of the allowable values, the possibility of consequences of an accident previously evaluated will not be increased.

Standard 2--Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated because, by maintaining the trip value at the midpoint of the allowable band, the proposed change is more restrictive. Since the change reduces the allowable values of the trip to a single value, which was part of the original safety analysis, the possibility of a new or different kind of accident from any accident previously evaluated will not be created.

Standard 3--Involve a significant reduction in a margin of safety.

The proposed change does not involve a significant reduction in a margin of safety because, by maintaining the trip value at the midpoint of the allowable band, the proposed change is more restrictive. By restricting the allowed operation of the Tank even further within the allowable trip values, the Unit does not experience as many possible accidents as before. Therefore, the change will not reduce the margin of safety.

2. The proposed amendment matches the guidance concerning the application of standards for determining whether or not a significant hazards consideration exists (51 FR 7751) by example:

   ii) A change that constitutes an additional limitation, restriction or control not presently included in the Technical Specifications: for example, a more stringent surveillance requirement.

E. SAFETY EVALUATION FOR THE AMENDMENT REQUEST

The proposed Technical Specification amendment will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR. The proposed change does not change or replace equipment or components important to safety. The change only limits the allowable values of the trip to a single value and is more restrictive by maintaining the trip value at the midpoint of the allowable band. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR will not be increased.

The proposed Technical Specification amendment will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR. The proposed change is more restrictive by maintaining the trip value at the midpoint of the allowable band. Since the change reduces the allowable values of the trip to a single value which was part of the original safety analysis, the possibility of a different accident or malfunction will not be created.

The proposed Technical Specification amendment will not reduce the margin of safety as defined in the basis for the Technical Specifications. The proposed change is more restrictive by maintaining the trip value at the midpoint of the allowable band. By restricting the allowed operation of the Tank even further within the allowable trip values, the Unit does not experience as many possible accidents as before. Therefore, the change will not reduce the margin of safety.

F. ENVIRONMENTAL IMPACT CONSIDERATION DETERMINATION

The proposed change request does not involve an unreviewed environmental question because operation of PVNGS Unit 2, in accordance with this change, would not:

1. Result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the staff's testimony to the Atomic Safety and Licensing Board; or

2. Result in a significant change in effluents or power levels; or

3. Result in matters not previously reviewed in the licensing basis for PVNGS which may have a significant environmental impact.

G. MARKED-UP TECHNICAL SPECIFICATION CHANGE PAGES

Limiting Conditions For Operation And Surveillance Requirements:

3/4 3-26

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

ESFA SYSTEM FUNCTIONAL UNIT	TRIP VALUES	ALLOWABLE VALUES
V. RECIRCULATION (RAS)		
A. Sensor/Trip Units		
Refueling Water Storage Tank - Low	$\geq$ 7.4% of Span	7.9 $\geq$ % of Span $\geq$ 6.9
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation System	Not Applicable	Not Applicable
VI. AUXILIARY FEEDWATER (SG-1)(AFAS-1)		
A. Sensor/Trip Units		
1. Steam Generator #1 Level - Low	$\geq$ 25.8% WR(4)	$\geq$ 25.3% WR(4)
2. Steam Generator Δ Pressure - SG2 > SG1	$\leq$ 185 psid	$\leq$ 192 psid
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
VII. AUXILIARY FEEDWATER (SG-2)(AFAS-2)		
A. Sensor/Trip Units		
1. Steam Generator #2 Level - Low	$\geq$ 25.8% WR(4)	$\geq$ 25.3% WR(4)
2. Steam Generator Δ Pressure - SG1 > SG2	$\leq$ 185 psid	$\leq$ 192 psid
B. ESFA System Logic	Not Applicable	Not Applicable
C. Actuation Systems	Not Applicable	Not Applicable
VIII. LOSS OF POWER		
A. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	$\geq$ 3250 volts	$\geq$ 3250 volts
B. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	2930 to 3744 volts with a 35-second maximum time delay	2930 to 3744 volts with a 35-second maximum time delay
IX. CONTROL ROOM ESSENTIAL FILTRATION	$\leq 2 \times 10^{-5}$ μCi/cc	$\leq 2 \times 10^{-5}$ μCi/cc

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ATTACHMENT 14

A. DESCRIPTION OF THE TECHNICAL SPECIFICATION AMENDMENT REQUEST

The proposed amendment is a number of administrative changes for the following Technical Specifications (T.S.):

Bases 3/4.3.1 and 3/4.3.2

1) page 3-2 remove Cycle 1 specific information no longer needed for Cycle 2

Bases 2.2.1

1) page 2-2 remove reference to CESSAR for description of the method of calculation for the trip variables for DNBR-Low and Local Power Density High trips and replace with the correct CE Topicals

2) page 2-3 update the latest revision used for calculating the PVNGS trip setpoint values

B. PURPOSE OF THE TECHNICAL SPECIFICATION

The purpose of T.S. 3.3.1 is to ensure that (1) the associated Engineered Safety Features Actuation action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

C. NEED FOR THE TECHNICAL SPECIFICATION AMENDMENT

The administrative changes are required to ensure clarity and conciseness. The change to Bases 3/4.3.1 removes information which pertained to Cycle 1 and is no longer valid for Cycle 2. The change to Bases 2.2.1 changes the source of the description of the method of calculation for the trip variables for DNBR-Low and Local Power Density High trips from the CESSAR to the correct CE Topicals and updates the T.S. to the latest revision of CEN - 286 (V), Rev 2.

D. BASIS FOR PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

1. The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety.

   A discussion of these standards, as they relate to the amendment request, follows:

   Standard 1--Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed changes are administrative in nature. They eliminate incorrect and superfluous information, thus ensuring that the Technical Specifications are concise and understandable. Therefore, the changes ensure that the possibility of an accident previously evaluated will not be increased.

   Standard 2--Create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed changes will not create the possibility of a new or different kind of accident previously evaluated because the proposed changes are administrative in nature. They eliminate incorrect and superfluous information thus ensuring that the Technical Specifications are concise and understandable. Therefore, the changes ensure that the possibility of a new or different kind of accident from any accident previously evaluated will not be created.

   Standard 3--Involve a significant reduction in a margin of safety.

   The proposed changes do not involve a significant reduction in a margin of safety because the proposed changes are administrative in nature. They eliminate incorrect and superfluous information thus ensuring that the Technical Specifications are concise and understandable. Therefore, the changes ensure that the margin of safety is maintained.

2. The proposed amendment matches the guidance concerning the application of standards for determining whether or not a significant hazards consideration exists (51 FR 7751) by example:

   (i) A purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

E. SAFETY EVALUATION FOR THE AMENDMENT REQUEST

The proposed Technical Specification amendment will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR. The proposed change does not change any equipment or components important to safety. The proposed changes are administrative in nature. They eliminate incorrect and superfluous information thus ensuring that the Technical Specifications are concise and understandable. Therefore, the changes ensure that the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR will not be increased.

The proposed Technical Specification amendment will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR. The proposed changes are administrative in nature. They eliminate incorrect and superfluous information, thus ensuring that the Technical Specifications are concise and understandable. Therefore, the changes ensure that the possibility of a different accident or malfunction will not be created.

The proposed Technical Specification amendment will not reduce the margin of safety as defined in the basis for the Technical Specifications. The proposed changes are administrative in nature. They eliminate incorrect and superfluous information thus ensuring that the Technical Specifications are concise and understandable. Therefore, the changes ensure that the margin of safety is maintained.

F. ENVIRONMENTAL IMPACT CONSIDERATION DETERMINATION

The proposed change request does not involve an unreviewed environmental question because operation of PVNGS Unit 2, in accordance with this change, would not:

1. Result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the staff's testimony to the Atomic Safety and Licensing Board; or

2. Result in a significant change in effluents or power levels; or

3. Result in matters not previously reviewed in the licensing basis for PVNGS which may have a significant environmental impact.

G. MARKED-UP TECHNICAL SPECIFICATION CHANGE PAGES

Limiting Conditions For Operation And Surveillance Requirements:

B 3/4 3-2 B 2-2
B 3/4 3-1 B 2-3

Limiting Safety System Settings for the Low DNBR, High Local Power Density, High Logarithmic Power Level, Low Pressurizer Pressure and High Linear Power Level trips, and Limiting Conditions for Operation on DNBR and kW/ft margin are specified such that there is a high degree of confidence that the specified acceptable fuel design limits are not exceeded during normal operation and design basis anticipated operational occurrences.

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 Coolant System components are designed to Section III, 1974 Edition, Summer 1975 Addendum, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of 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.

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and Reactor Coolant System are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

The DNBR - Low and Local Power Density - High are digitally generated trip setpoints based on Safety Limits of 1.231 and 21 kW/ft, respectively. Since these trips are digitally generated by the Core Protection Calculators, the trip values are not subject to drifts common to trips generated by analog type equipment. The Allowable Values for these trips are therefore the same as the Trip Setpoints.

To maintain the margins of safety assumed in the safety analyses, the calculations of the trip variables for the DNBR - Low and Local Power Density - High trips include the measurement, calculational and processor uncertainties and dynamic allowances as defined in ~~CESSAR System 80 applicable system descriptions and safety analyses.~~ the latest applicable revision of CEN-305-P, "Functional Design Requirements for a Core Protection Calculator" and CEN-304-P, "Functional Design Requirements for a Control Element Assembly Calculator."

REACTOR TRIP SETPOINTS (Continued)

The methodology for the calculation of the PVNGS trip setpoint values, plant protection system, is discussed in the CE Document No. CEN-286(V), Rev. 2, dated ~~July 31, 1984~~. August 29, 1986.

Manual Reactor Trip

The Manual reactor trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Variable Overpower Trip

A reactor trip on Variable Overpower is provided to protect the reactor core during rapid positive reactivity addition excursions. This trip function will trip the reactor when the indicated neutron flux power exceeds either a rate limited setpoint at a great enough rate or reaches a preset ceiling. The flux signal used is the average of three linear subchannel flux signals originating in each nuclear instrument safety channel. These trip setpoints are provided in Table 2.2-1.

Logarithmic Power Level - High

The Logarithmic Power Level - High trip is provided to protect the integrity of fuel cladding and the Reactor Coolant System pressure boundary in the event of an unplanned criticality from a shutdown condition. A reactor trip is initiated by the Logarithmic Power Level - High trip unless this trip is manually bypassed by the operator. The operator may manually bypass this trip when the THERMAL POWER level is above 10^{-4}% of RATED THERMAL POWER; this bypass is automatically removed when the THERMAL POWER level decreases to 10^{-4}% of RATED THERMAL POWER.

Pressurizer Pressure - High

The Pressurizer Pressure - High trip, in conjunction with the pressurizer safety valves and main steam safety valves, provides Reactor Coolant System protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is below the nominal lift setting of the pressurizer safety valves and its operation minimizes the undesirable operation of the pressurizer safety valves.

Pressurizer Pressure - Low

The Pressurizer Pressure - Low trip is provided to trip the reactor and to assist the Engineered Safety Features System in the event of a decrease in Reactor Coolant System inventory and in the event of an increase in heat

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3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the reactor protective and Engineered Safety Features Actuation Systems instrumentation and bypasses ensures that (1) the associated Engineered Safety Features Actuation action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses.

Response time testing of resistance temperature devices, which are a part of the reactor protective system, shall be performed by using in-situ loop current test techniques or another NRC approved method.

The Core Protection Calculator (CPC) addressable constants are provided to allow calibration of the CPC system to more accurate indications of power level, RCS flow rate, axial flux shape, radial peaking factors and CEA deviation penalties. Administrative controls on changes and periodic checking of addressable constant values (see also Technical Specifications 3.3.1 and 6.8.1) ensure that inadvertent misloading of addressable constants into the CPCs is unlikely.

The design of the Control Element Assembly Calculators (CEAC) provides reactor protection in the event one or both CEACs become inoperable. If one CEAC is in test or inoperable, verification of CEA position is performed at least every 4 hours. If the second CEAC fails, the CPCs in conjunction with plant Technical Specifications will use DNBR and LPD penalty factors and increased DNBR and LPD margin to restrict reactor operation to a power level that will ensure safe operation of the plant. If the margins are not maintained, a reactor trip will occur.

The value of the DNBR in Specification 2.1 is conservatively compensated for measurement uncertainties. Therefore, the actual RCS total flow rate determined by the reactor coolant pump differential pressure instrumentation or by calorimetric calculations does not have to be conservatively compensated for measurement uncertainties.

~~An analysis was done to specify a minimum power level below which an additional power reduction is unnecessary even if there is a CEA misalignment with CEACs out of service.~~

REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

The analysis determined a Power Operating Limit (POL) power and assumed a CEA misalignment occurred from this power level. The power penalty factor that would accommodate changes in radial peaks and one hour xenon redistribution that would occur if there were a CEA misalignment with CEACs out of service. The quotient of the POL power and the CEA misalignment Power Penalty factor is the maximum power (50% power) at which DNBR SAFDL violation will occur even if there is a CEA misalignment from POL conditions. Below this power, extra thermal margin will be available to the plant. Thus, for CEA misalignment, power reduction below this limiting power is unnecessary.

The lowest core power for a POL was calculated to be 70% of rated power. This was based on the following worst COLSS fluid conditions.

High Temperature	:	580°F
Low Pressure	:	1785 psia
ASI	:	-.3
Underflow fraction	:	0.865
Low Flow	:	95% of full flow
High Radial Peak	:	1.70 (Bank 5+4+PLR; PDIL = 40% Power)

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. The response times in Table 3.3-2 are made up of the time to generate the trip signal at the detector (sensor response time) and the time for the signal to interrupt power to the CEA drive mechanism (signal or trip delay time). ~~The response times are taken from the sequence of events Tables in Section 15 of CESSAR.~~

Response time may be demonstrated by any series of sequential, overlapping, or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) in place, onsite, or offsite test measurements or (2) utilizing replacement sensors with certified response times.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that: (1) the radiation levels are continually measured in the areas served by the