NPF-38-08

DESCRIPTION AND SAFETY ANALYSIS
OF PROPOSED CHANGE NPF-38-08

This is a request to modify the Technical Specifications to remove the Core Protection Calculator (CPC) Type I and Type II Addressable Constants, as approved by the NRC Core Performance Branch in April, 1985.

Existing Specification

See Attachment "A"

Proposed Specification

See Attachment "B"

Description

The proposed change will delete Technical Specification 2.2.2, "Core Protection Calculator Addressable Constants"; delete Table 2.2-2, which provides a listing of the CPC Type I and Type II Addressable Constants; and delete the associated Bases. The proposed change will also revise the appropriate page of the Index, delete the reference to specification 2.2.2 from Notation (9) of Table 4.3-1, and delete the note in Administrative Control 6.8.1(g).

The addressable constants of the Core Protection Calculators provide a mechanism to incorporate reload dependent parameters and calibration constants to the CPC software so that the CPC core model is maintained current with changing core configurations and operating characteristics. The CPC software has been designed with automatic acceptable input checks against limits that are specified by the CPC functional design specifications. Therefore, inclusion of the addressable constants and the software limit values in the Technical Specifications (2.2.2 and Table 2.2-2) is redundant. Furthermore, inclusion of addressable constant values in the Technical Specifications that are more restrictive than the software limit values does not result in additional safety benefit because existing LCOs (e.g. 3.2.3, 3.3.1) either provide adequate assurance that CPC calculated values are accurate, or prohibit operation with non-conservative addressable constant values.

Proper administrative control procedures are available to assure that correct values of addressable constants are entered by the operator. In addition, CPC software changes involving addressable constants or software limit values are made and tested under NRC approved software change procedures and are available for NRC review.

Safety Analysis

The proposed changes described above shall be deemed to involve a significant hazards consideration if there is a positive finding in any of the following areas:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of any accident previously evaluated?

   Response: No.

   This change eliminates redundant administrative requirements concerning the CPC addressable constants. The function of these requirements is already implemented by the allowable value checks in the CPC software. Changes to the addressable constants are accomplished through strict administrative procedures. Therefore, the proposed changes will not involve an increase in the probability or consequences of any accident previously evaluated.

2. Will operation of the facility in accordance with the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   This change involves an elimination of redundant administrative requirements. The analyses in Chapter 15 of the Waterford 3 FSAR continue to bound all events where the CPCs may be challenged. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

   Response: No.

   Safety margin is governed by LCOs (e.g. 3.2.3, 3.3.1) independently of the addressable constant limits of Table 2.2-2. Administrative procedures involving the CPC addressable constants ensure that the CPC core model is calibrated to current plant conditions. Therefore, the proposed change will not involve a reduction in a margin of safety.

The commission has provided guidance concerning the application of standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards considerations. Example (i) relates to 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. Example (iv) relates to a relief granted upon demonstration of acceptable operation from an operating restriction that was imposed because acceptable operation was not yet demonstrated. This assumes that the operating restriction and the criteria to be applied to a request for relief have been established in a prior review and that it is justified in a satisfactory way that the criteria have been met.

In this case, the proposed change is similar to both Example (i) and Example (iv) in that deletion of Technical Specification 2.2.2, Table 2.2-2 and modifications to the related pages are purely administrative changes, and are also relief granted upon demonstration of acceptable operation from an operating restriction that was imposed because acceptable operation had not yet been demonstrated.

Conceptually, the addressable constant reasonability checks are the equivalent of the limits of an adjustable potentiometer in the conventional analog hard-wired type protection system. Potentiometer limits are not specified in the Technical Specifications of analog plants, as doing so would make no contribution to plant safety. The addressable constants are basically calibration constants used to assure that the CPC calculations of core parameters accurately reflect actual plant conditions. The proposed change may therefore be considered to achieve consistency throughout the Technical Specifications in that it removes a listing of calibration constants which is redundant in purpose and is not provided for any other plant system.

Removal of the listing of the CPC addressable constants and the allowable ranges is a relief from an operating restriction that was imposed by the NRC CPC Review Task Force. The addressable constants Technical Specification was imposed on the first CPC Plant because this system was the first application of a digital computer based reactor protection system. Subsequent operational experience with the CPC system by several plants has demonstrated acceptable operation. Relief from this administrative restriction has been approved following several meetings between the utilities with CPC equipped plants and the NRC Core Performance Branch, which included members of the CPC Review Task Force. The criteria applied to the relief from this operating restriction have been established and there is satisfactory justification that they have been met. The NRC Core Performance Branch has issued a draft Safety Evaluation Report (concerning the removal of the addressable constants Technical Specification) which provides this justification.

Safety and Significant Hazards Determination

Based on the above safety analysis, it is concluded that: (1) the proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.92; and (2) there is a reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

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ATTACHMENT A

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SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

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2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

2.2.2 Core Protection Calculator Addressable Constants shall be within the limits specified in Table 2.2-2.

APPLICABILITY: As shown for Core Protection Calculators in Table 3.3-1

ACTION:

With a Core Protection Calculator Addressable Constant outside the limits of Table 2.2-2, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status.

2.2.2 CPC ADDRESSABLE CONSTANTS

The Core Protection Calculator (CPC) addressable constants are provided to allow calibration of the CPC system to more accurate indications such as calorimetric measurements for power level and RCS flowrate and incore detector signals for axial flux shape, radial peaking factors and CEA deviation penalies. Administrative controls on changes and periodic checking of addressable constant values (see also Technical Specification 4.3.1.1, footnote 9 to Table 4.3-1, and Specification 6.8.1) ensure that inadvertent misloading of addressable constants into the CPCs is unlikely.

The upper and lower limits of the CPC addressable constants in Table 2.2-2 are based on the validation limits of the computer software that uses the constants, or the NRC staff's analysis of the constants. Any modifications which are made to the core protection calculator software (including changes of algorithms and addressable constants or fuel cycle-specific data) shall be within these limits and be made in accordance with "CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P, Revision 2 and Supplement 1-P, Revision 01 or another NRC-approved procedure on CPC software modifications.

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

*With the reactor trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

(1) Each startup or when required with the reactor trip breakers closed and the CEA drive system capable of rod withdrawal, if not performed in the previous 7 days.

(2) Heat balance only (CHANNEL FUNCTIONAL TEST not included), above 15% of RATED THERMAL POWER: adjust the Linear Power Level signals and the CPC addressable constant multipliers to make the CPC ΔT power and CPC nuclear power calculations agree with the calorimetric calculation if absolute difference is greater than 2%. During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) Above 15% of RATED THERMAL POWER, verify that the linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine the shape annealing matrix elements and the Core Protection Calculators shall use these elements.

(6) This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

(7) Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation or by calorimetric calculations and if necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty is included in the BERR1 term in the CPC and is equal to or greater than 4%.

(8) Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by calorimetric calculations.

(9) The monthly CHANNEL FUNCTIONAL TEST shall include verification that the correct values of addressable constants are installed in each OPERABLE CPC per Specification 2.2.2.

(10) At least once per 18 months and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include independent verification of the undervoltage trip function and the shunt trip function.

TABLE 2.2-2

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE I ADDRESSABLE CONSTANTS

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION	LOWER LIMITS	UPPER LIMITS
60	FC1	Core coolant mass flow rate calibration constant	.8	1.15
61	FC2	Core coolant mass flow rate calibration constant	0.0	0.0
62	CEANOP	CEAC/RSPT inoperable flag	*	*
63	TR	Azimuthal tilt allowance	1.02	1.4
64	TPC	Thermal power calibration constant	.7	1.3
65	KCAL	Neutron flux power calibration constant	0.0	2.0
66	DNBRPT	DNBR pretrip setpoint	1.25	5.0
67	LPDPT	Local power density pretrip setpoint	10.0	20.0

II. TYPE II ADDRESSABLE CONSTANTS

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION	LOWER LIMITS	UPPER LIMITS
68	BERR0	Thermal power uncertainty bias	**	**
69	BERR1	Power uncertainty factor used in DNBR calculation	**	**
70	BERR2	Power uncertainty bias used in DNBR calculation	**	**

*The CEAC/RSPT inoperable flag must have the value 0, 1, 2, or 3

**These shall be those established in accordance with CEN-197(C)-P, CPC/CEAC Software Modification for Waterford 3, March 1982.

Table 2.2-2 (Continued)

II. TYPE II ADDRESSABLE CONSTANTS

			LOWER LIMITS	UPPER LIMITS
71	BERR2	Power uncertainty factor used in DNBR calculation	**	**
72	BERR4	Power uncertainty bias used in local power density calculation	**	**
73	EOL	End of life flag	**	**
74	ARM1	Multiplier for planar radial peaking factor	**	**
75	ARM2	Multiplier for planar radial peaking factor	**	**
76	ARM3	Multiplier for planar radial peaking factor	**	**
77	ARM4	Multiplier for planar radial peaking factor	**	**
78	ARM5	Multiplier for planar radial peaking factor	**	**
79	ARM6	Multiplier for planar radial peaking factor	**	**
80	ARM7	Multiplier for planar radial peaking factor	**	**
81	SC11	Shape annealing correction factor	**	**
82	SC12	Shape annealing correction factor	**	**
83	SC13	Shape annealing correction factor	**	**

Table 2.2-2 (Continued)

II. TYPE II ADDRESSABLE CONSTANTS			LOWER LIMITS	UPPER LIMITS
84	SC21	Shape annealing correction factor	**	**
85	SC22	Shape annealing correction factor	**	**
86	SC23	Shape annealing correction factor	**	**
87	SC31	Shape annealing correction factor	**	**
88	SC32	Shape annealing correction factor	**	**
89	SC33	Shape annealing correction factor	**	**
90	PFMLTD	DNBR penalty factor correction multiplier	**	**
91	PFMLTL	LPD penalty factor correction multiplier	**	**
92	ASM2	Multiplier for CEA shadowing factor	**	**
93	ASM3	Multiplier for CEA shadowing factor	**	**
94	ASM4	Multiplier for CEA shadowing factor	**	**
95	ASM5	Multiplier for CEA shadowing factor	**	**
96	ASM6	Multiplier for CEA shadowing factor	**	**
97	ASM7	Multiplier for CEA shadowing factor	**	**
98	CORR1	Temperature shadowing correction factor multiplier	**	**
99	BPPCC1	Boundary point power correlation coefficient	**	**
100	BPPCC2	Boundary point power correlation coefficient	**	**

Table 2.2-2 (Continued)

II. TYPE II ADDRESSABLE CONSTANTS

			LOWER LIMITS	UPPER LIMITS
101	BPPCC3	Boundary point power correlation coefficient	**	**
102	BPPCC4	Boundary point power correlation coefficient	**	**
103	RPCLIM	Reactor power cutback time limit	**	**

SAFETY LIMIT VIOLATION (Continued)

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Senior Vice President-Nuclear Operations and the SRC shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems, or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the SRC and the Senior Vice President-Nuclear Operations within 14 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978 and those required for implementing the requirements of NUREG-0737.

b. Refueling operations.

c. Surveillance and test activities of safety-related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. Modification of Core Protection Calculator (CPC) Addressable Constants, including independent verification of modified constants.

NOTE: Modification to the CPC addressable constants based on information obtained through the Plant Computer - CPC data link shall not be made without prior approval of the PORC.

h. Administrative procedures implementing the overtime guidelines of Specification 6.2.2f., including provisions for documentation of deviations.

i. PROCESS CONTROL PROGRAM implementation.

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ATTACHMENT B

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

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2.2 LIMITING SAFETY SYSTEM SETTINGS		
2.2.1	REACTOR TRIP SETPOINTS	B 2-2
~~2.2.2~~	~~CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS~~	~~B 2-7~~

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

2.2.2 Core Protection Calculator Addressable Constants shall be within the limits specified in Table 2.2-2.

APPLICABILITY: As shown for Core Protection Calculators in Table 3.3-1.

ACTION: DELETE

With a Core Protection Calculator Addressable Constant outside the limits of Table 2.2-2, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status.

2.2.2 CPC ADDRESSABLE CONSTANTS

The Core Protection Calculator (CPC) addressable constants are provided to allow calibration of the CPC system to more accurate indications such as calorimetric measurements for power level and RCS flowrate and incore detector signals for axial flux shape, radial peaking factors and CEA deviation penalies. Administrative controls on changes and periodic checking of addressable constant values (see also Technical Specification 4.3.1.1, footnote 9 to Table 4.3-1, and Specification 6.8.1) ensure that inadvertent misloading of addressable constants into the CPCs is unlikely.

The upper and lower limits of the CPC addressable constants in Table 2.2-2 are based on the validation limits of the computer software that uses the constants, or the NRC staff's analysis of the constants. Any modifications which are made to the core protection calculator software (including changes of algorithms and addressable constants or fuel cycle-specific data) shall be within these limits and be made in accordance with "CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P, Revision 2 and Supplement 1-P, Revision 01 or another NRC-approved procedure on CPC software modifications.

DELETE

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

*With the reactor trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

(1) Each startup or when required with the reactor trip breakers closed and the CEA drive system capable of rod withdrawal, if not performed in the previous 7 days.

(2) Heat balance only (CHANNEL FUNCTIONAL TEST not included), above 15% of RATED THERMAL POWER: adjust the Linear Power Level signals and the CPC addressable constant multipliers to make the CPC ΔT power and CPC nuclear power calculations agree with the calorimetric calculation if absolute difference is greater than 2%. During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) Above 15% of RATED THERMAL POWER, verify that the linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine the shape annealing matrix elements and the Core Protection Calculators shall use these elements.

(6) This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

(7) Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation or by calorimetric calculations and if necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty is included in the BERR1 term in the CPC and is equal to or greater than 4%.

(8) Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by calorimetric calculations.

(9) The monthly CHANNEL FUNCTIONAL TEST shall include verification that the correct values of addressable constants are installed in each OPERABLE CPC. ~~per Specification 2.2.2.~~

(10) At least once per 18 months and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include independent verification of the undervoltage trip function and the shunt trip function.

TABLE 2.2-2

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II. TYPE II ADDRESSABLE CONSTANTS

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION	LOWER LIMITS	UPPER LIMITS
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*The CEAC/RSPT inoperable flag must have the value 0, 1, 2, or 3

**These shall be those established in accordance with CEN-197(C)-P, CPC/CEAC Software Modification for Waterford 3, March 1982.

Table 2.2-2 (Continued)

II. TYPE II ADDRESSABLE CONSTANTS

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83	SC13	Shape annealing correction factor	**	**

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Table 2.2-2 (Continued)

II. TYPE II ADDRESSABLE CONSTANTS

			LOWER LIMITS	UPPER LIMITS
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86	SC23	Shape annealing correction factor	**	**
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94	ASM4	Multiplier for CEA shadowing factor	**	**
95	ASM5	Multiplier for CEA shadowing factor	**	**
96	ASM6	Multiplier for CEA shadowing factor	**	**
97	ASM7	Multiplier for CEA shadowing factor	**	**
98	CORR1	Temperature shadowing correction factor multiplier	**	**
99	BPPCC1	Boundary point power correlation coefficient	**	**
100	BPPCC2	Boundary point power correlation coefficient	**	**

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Table 2.2-2 (Continued)

II. TYPE II ADDRESSABLE CONSTANTS			LOWER LIMITS	UPPER LIMITS
101	BPPCC3	Boundary point power correlation coefficient	**	**
102	BPPCC4	Boundary point power correlation coefficient	**	**
103	RPCLIM	Reactor power cutback time limit	**	**

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SAFETY LIMIT VIOLATION (Continued)

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Senior Vice President-Nuclear Operations and the SRC shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems, or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the SRC and the Senior Vice President-Nuclear Operations within 14 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978 and those required for implementing the requirements of NUREG-0737.

b. Refueling operations.

c. Surveillance and test activities of safety-related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. Modification of Core Protection Calculator (CPC) Addressable Constants, including independent verification of modified constants.

NOTE: Modification to the CPC addressable constants based on information obtained through the Plant Computer - CPC data link shall not be made without prior approval of the PORC.

h. Administrative procedures implementing the overtime guidelines of Specification 6.2.2f., including provisions for documentation of deviations.

i. PROCESS CONTROL PROGRAM implementation.

NPF-38-09

DESCRIPTION AND SAFETY ANALYSIS
OF PROPOSED CHANGE NPF-38-09

This is a request to modify the Technical Specifications to incorporate the results of the Waterford 3 COLSS Out-of-Service Analysis.

Existing Specification

See Attachment "A"

Proposed Specification

See Attachment "B"

Description

The proposed change will revise Technical Specifications 3.2.1, "Linear Heat Rate", 3.2.4, "DNBR Margin", 3.2.7, "Axial Shape Index", and the associated Bases. The proposed change will also add Figure 3.2-1a "Allowable Peak Linear Heat Rate vs Tc for COLSS Out-of-Service"; delete Figure 3.2-2 "DNBR Margin Operating Limit Based on COLSS" and replace it with "DNBR Margin Operating Limit Based on CPC's (COLSS Out-of-Service, CEACs Operable)"; Revise Figure 3.2-3 "DNBR Margin Operating Limit Based on CPC's (COLSS Out-of-Service, CEAC Inoperable)"; and revise Action 6.b.1 of Table 3.3-1.

Waterford 3 operators use the Core Operating Limit Supervisory System (COLSS) to help monitor linear heat rate, DNBR margin, and axial shape index as presently required by Technical Specifications 3.2.1, 3.2.4 and 3.2.7. However, whenever COLSS is out of service, the operators must use the Core Protection Calculators (CPC) to perform the same function. Since CPC uses ex-core detectors and is required to take action during certain transients, its uncertainties and margins are more limiting than those of COLSS.

When CPC is being used to help monitor LCOs, the extra conservatisms built into CPC for transient protection are not required and thus can be removed. In order to not affect CPC's transient protection, these conservatisms will be credited in the "COLSS Out-of-Service" limits of Technical Specifications 3.2.1, 3.2.4, and 3.2.7 rather than removed from CPC. The COLSS Out-of-Service Margin Improvement Program performed for Waterford 3 Cycle 1 has determined the amount of credit available and the adjustment to these Technical Specifications that would be available to take advantage of this credit.

The proposed Technical Specification changes, as discussed above, result from taking advantage of conservatisms in the CPC monitoring of linear heat rate and DNBR margin. These conservatisms are discussed below.

1. Reduced Modeling Uncertainty Over a Narrower LCO ASI Band -

   The CPC modeling uncertainty involves the error in CPC's ability to determine a hot pin axial power distribution (for DNBR) and 3-D peaking (for linear heat rate) based on excore detector signals and CEA positions. The uncertainty could be sensitive to the ASI range considered since more unusual axial shapes are normally more difficult for the CPC to model. Therefore, the ASI range was reduced to obtain additional credit for LHR and DNBR due to the difference between the uncertainty over the LSSS range (± .6 ASI) and the uncertainty over the reduced LCO range (± .3 ASI).

2. Credit for Transient Offset Terms -

CPC contains power penalties which compensate for potential nonconservatisms during certain rapid transients. The CPC must contain these penalties in order to provide conservative transient protection. However, the penalties are not required to assure conservative LHR and DNBR measurements during operation within the LCOs. Therefore, the penalties can be credited in the limits monitored by the CPC when COLSS is out of service. The penalties do, however, remain in the CPC in order to maintain conservative transient protection.

3. Credit for Additive Power Measurement Uncertainty -

CPC includes a penalty to cover the uncertainty in the secondary calorimetric power measurement which is used to calibrate CPC power values. This uncertainty is larger at lower power levels. Since the CPCs for Waterford 3 Cycle 1 have a single power independent uncertainty factor, the maximum (low power) value is implemented. For monitoring purposes, conservatism at lower power levels is not essential since the core cannot approach DNBR and LHR limits at low power except when a transient occurs. The minimum difference between the required uncertainty at high power and the implemented uncertainty can be credited in the DNBR and LHR limits when using CPC for monitoring.

4. Credit for Conservatism of Neutron Flux Power Relative to Thermal Power -

Neutron flux power and thermal power are both used in CPC in order to assure conservatism during any CPC design basis transient. Neutron flux power provides faster response due to its use of excore detectors while thermal power provides more accurate response during asymmetric CEA transients where neutron flux power is decalibrated by the asymmetry. For monitoring, conservatism during asymmetric and/or rapid transients is not required. Therefore, either power value will be sufficient. Extra conservatism in neutron flux power relative to thermal power can be credited in the DNBR and LHR limits when CPC is used for monitoring.

5. Credit for Dynamic Pressure Uncertainty in CPC -

CPC for Waterford 3 Cycle 1 includes a penalty applied to measured pressure in the conservative direction to assure conservative DNBR during operation within the LCOs. Therefore the excess conservatism that it represents can be credited in the DNBR limit when CPC is used for monitoring.

6. Credit for the Update Penalty Factor in CPC -

CPC includes a fast running UPDATE program which modifies the DNBR calculated by the detailed STATIC program to account for changes in input parameters (e.g. pressure, temperature, CEA position, excore detector signals) which may occur between executions of STATIC. Since the UPDATE program uses approximate derivatives of various parameters to determine the DNBR, a penalty factor is applied to compensate for any uncertainty in the UPDATE calculation for any transient. The use of CPC for LCO monitoring would not require conservative UPDATE calculations since the execution period of the more accurate STATIC program would be sufficient to satisfy monitoring requirements. Therefore, the penalty that is always applied to the CPC DNBR calculation to compensate for the uncertainty in the UPDATE program can be credited in the DNBR limit when CPC is used for monitoring.

The conservatisms identified above provide approximately 10% linear heat rate and 12 % DNBR power margin credits. These credits have been translated into COLSS Out-of-Service LCO adjustments in the proposed Technical Specification changes.

These proposed changes are similar to those changes (NPF-10-52) submitted by the Southern California Edison Company (SCE) on August 7, 1984 in a letter to Mr. H. R. Denton. The changes were approved by the NRC and Amendment No. 32 to the SCE Operating License was issued in a letter from the NRC to K. P. Baskin and J. C. Holcombe on March 1, 1985.

Safety Analysis

The proposed change described above shall be deemed to involve a significant hazards consideration if there is a positive finding in any of the following areas:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of any accident previously evaluated?

   Response: No.

   The proposed Technical Specification changes do not affect the CPC transient protection. The conservatisms are not taken from the CPC but rather are credited in the COLSS out-of-service limits of Technical Specification 3.2.1, 3.2.4 and 3.2.7. Therefore, the proposed changes will not involve an increase in the probability or consequence of any accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed changes in no way affect the steady state or transient operation of the CPC. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

   Response: No.

The proposed change will not affect the margin of safety because the CPC operation is in no way altered. Plant Operators use the COLSS to help monitor linear heat rate, DNBR margin, and axial shape index as required by the Technical Specifications. However, when COLSS is out-of-service the operators must use the CPC to perform the same function. Since the CPC uses ex-core detectors and is required to take action during certain transients, the uncertainties and margins are more limiting than that of COLSS.

When the CPC is used to help monitor LCOs, the extra conservatisms built into the CPC for transient protection are not required and thus can be removed. In order to no affect the CPC transient protection, these conservatisms will be credited in the "COLSS Out-of-Service" limits of Technical Specifications 3.2.1, 3.2.4 and 3.2.7 rather than removed from the CPC. Therefore, the proposed changes will not involve any reduction in a margin of safety.

The Commission has provided guidance concerning the application of standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards considerations. Example (vi) relates to a change which may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria specified in the Standard Review Plan.

In this case the proposed change described above is similar to Example (vi) in that revising Technical Specifications 3.2.1, 3.2.4 and 3.2.7; revising Figure 3.2.3 and Action 6.6.1 of Table 3.3-1; replacing Figure 3.2-2; and revising the associated Bases is clearly within all acceptable criteria specified by the NRC for digital plant protection systems in general and the CPC in particular. Additionally, the changes provide for less impact than allowed by Example (vi) because no safety margins are reduced.

These changes are similar to those (NPF-10-52) submitted by SCE on August 7, 1984 and approved by the NRC. Amendment No. 32 to the SCE Operating License was issued in a letter from the NRC (G. W. Knighton) to K.P. Baskin and J. C. Holcombe on March 1, 1985.

Safety and Significant Hazards Determination

Based on the above safety analysis, it is concluded that: (1) the proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.92; and (2) there is a reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

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ATTACHMENT A

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 the limits shown on Figure 3.2-1.

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

ACTION:

With the linear heat rate exceeding its limits, 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 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. Be in at least HOT STANDBY 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 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 linear heat rate, as indicated on all OPERABLE Local Power Density channels, is within the limits shown on Figure 3.2-1.

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 kW/ft.

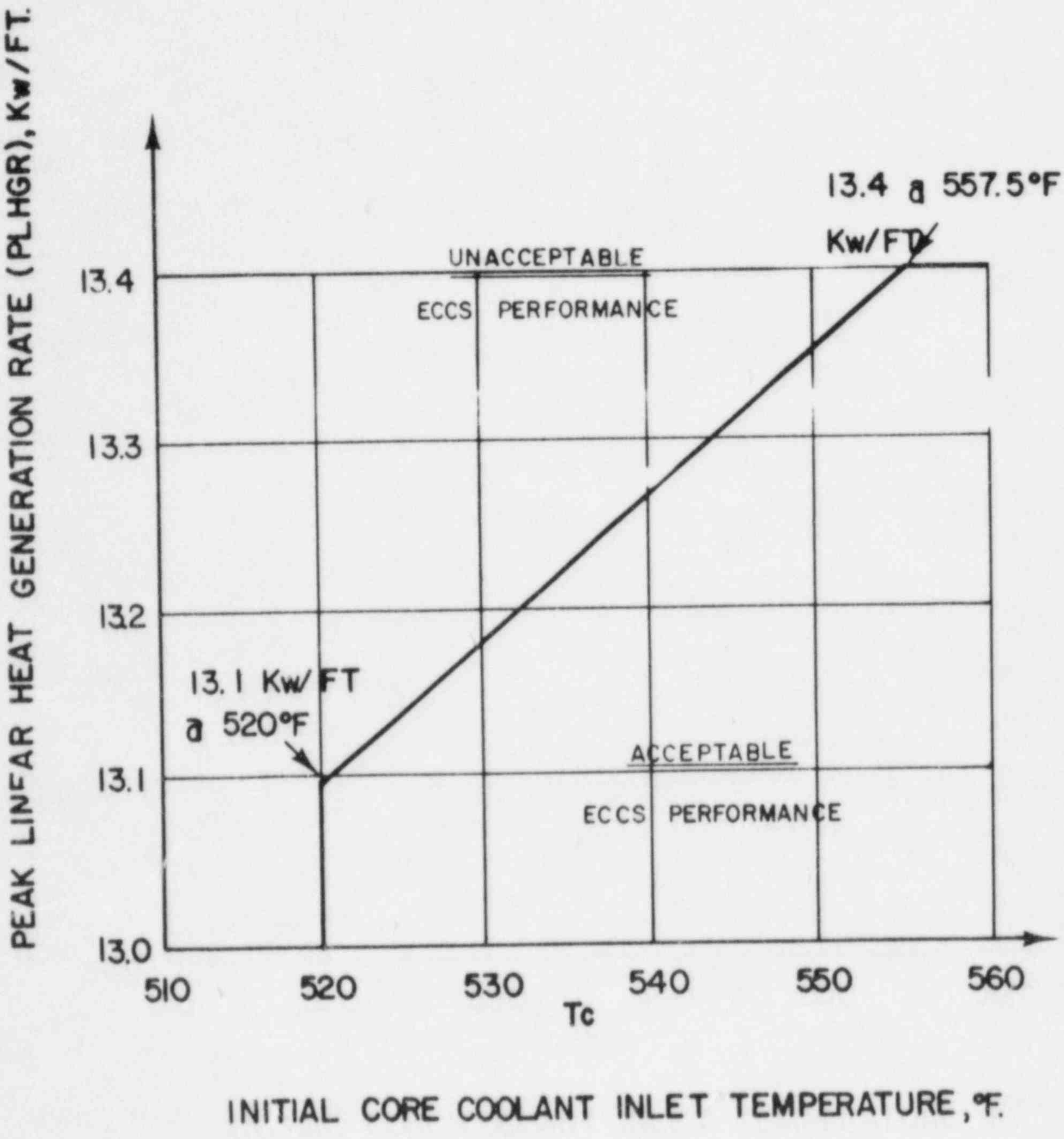

FIGURE 3.2-1

ALLOWABLE PEAK LINEAR HEAT RATE VS Tc

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 3.2-2 or 3.2-3, as applicable.

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 exceeding the DNBR limit, within 15 minutes initiate 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.

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 all OPERABLE DNBR channels, is within the limit shown on Figure 3.2-3.

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.

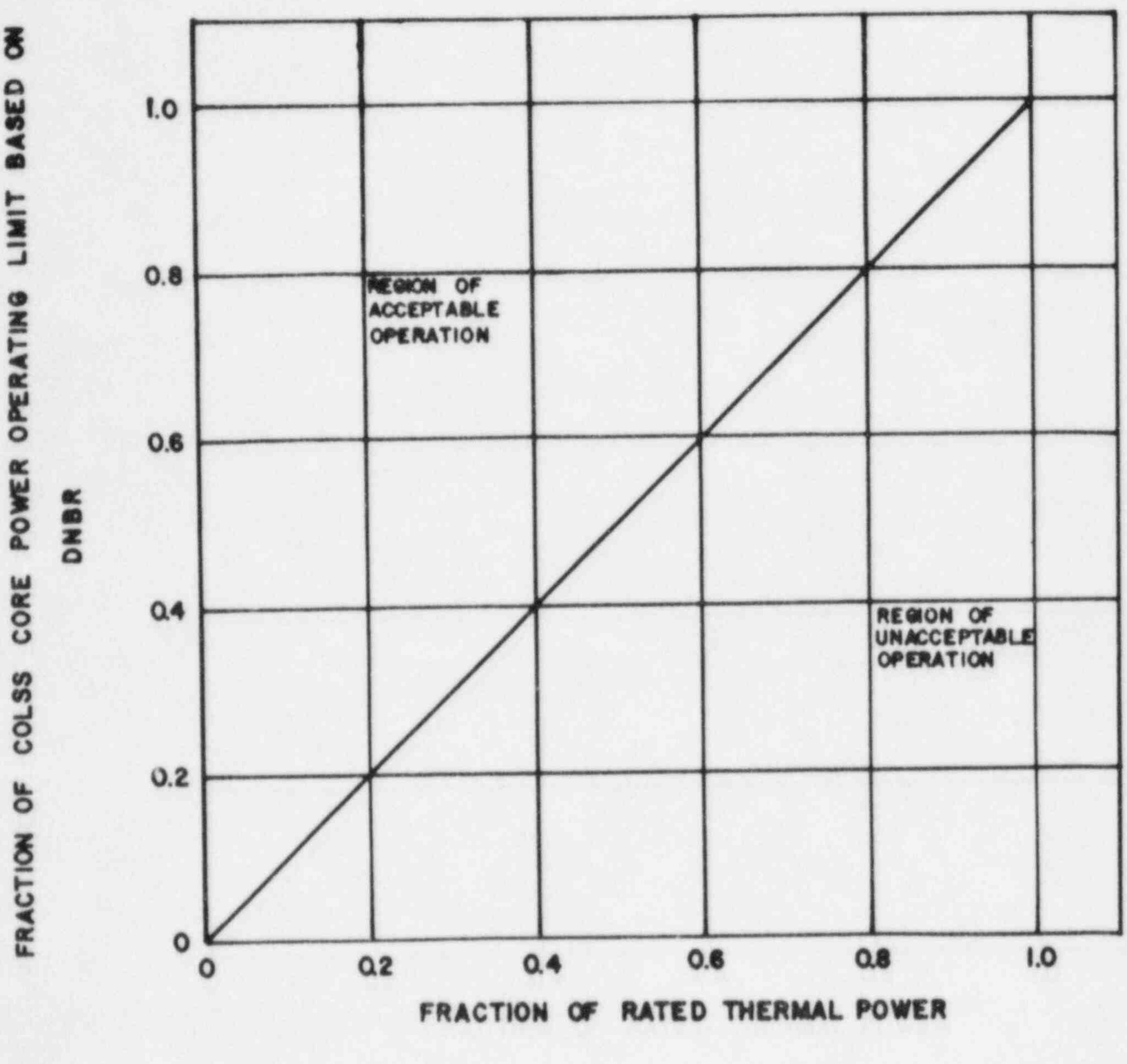

FIGURE 3.2-2

DNBR MARGIN OPERATING LIMIT BASED ON COLSS

COLSS OUT OF SERVICE DNBR LIMIT LINE

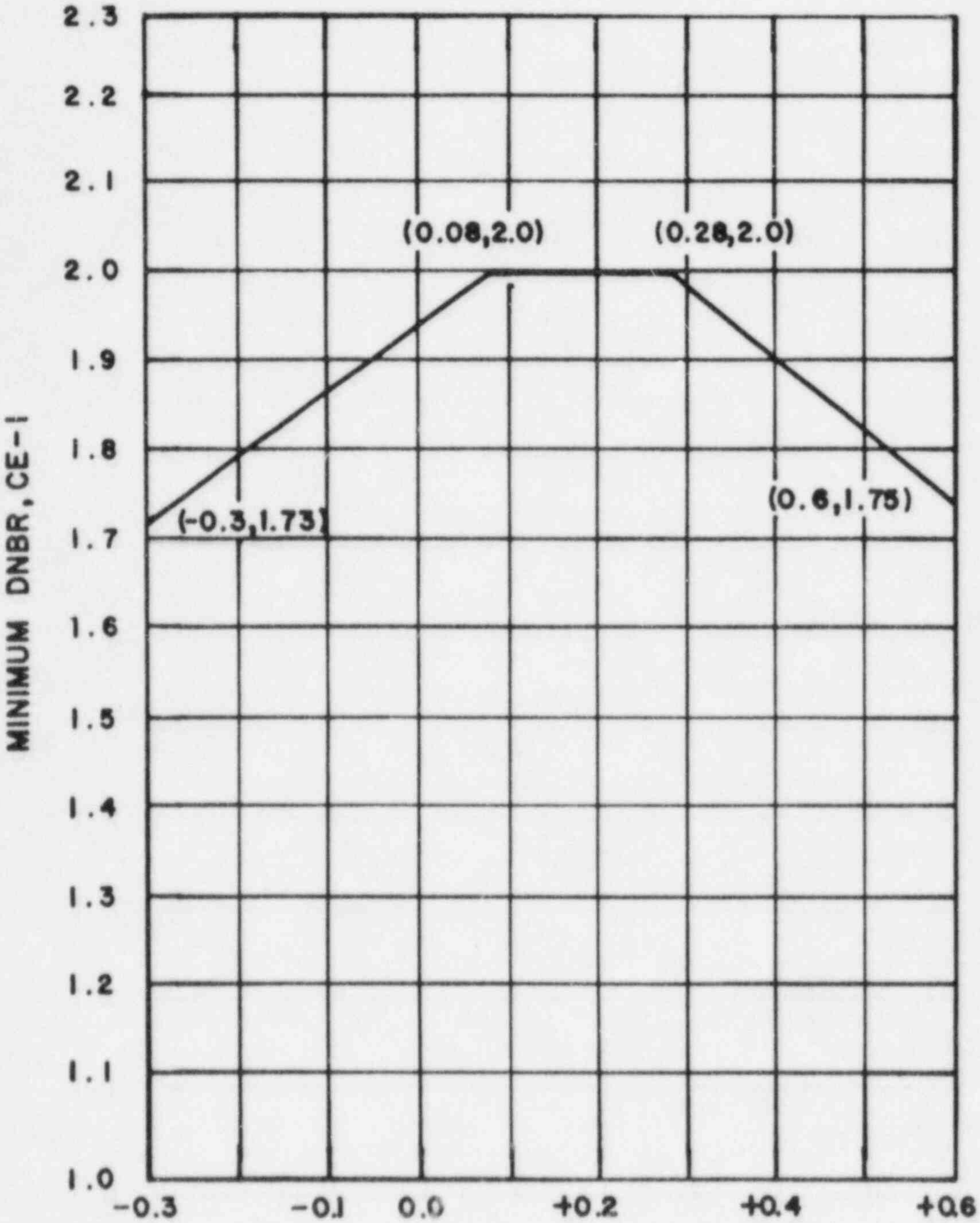

FIGURE 3.2-3

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

3/4.2.7 AXIAL SHAPE INDEX

LIMITING CONDITION FOR OPERATION

3.2.7 The AXIAL SHAPE INDEX (ASI) shall be maintained within the following limits:

a. COLSS OPERABLE
$-0.23 \leq \text{ASI} \leq +0.50$

b. COLSS OUT OF SERVICE (CPC)
$-0.15 \leq \text{ASI} \leq +0.50$

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

ACTION:

With the AXIAL SHAPE INDEX outside its above limits, restore the AXIAL SHAPE INDEX to within its limit within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.7 The AXIAL SHAPE INDEX shall be determined to be within its limit at least once per 12 hours using the COLSS or any OPERABLE Core Protection Calculator channel.

---

*See Special Test Exception 3.10.2.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

2.	Pressurizer Pressure - High	Pressurizer Pressure - High Local Power Density - High DNBR - Low
3.	Containment Pressure - (RPS) High	Containment Pressure - High Containment Pressure - High (ESF)
4.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS 1 and 2)
5.	Steam Generator Level	Steam Generator Level - Low Steam Generator Level - High Steam Generator ΔP (EFAS)
6.	Core Protection Calculator	Local Power Density - High DNBR - Low

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 those required by the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the reactor trip breakers of the inoperable channel are placed in the tripped condition within 1 hour; otherwise, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed 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 7 inches (indicated position) of all other CEAs in its group.

b. With both CEACs inoperable, operation may continue provided that:

1. Within 1 hour the margins required by Specification 3.2.1 and 3.2.4 are increased and maintained at a value equivalent to greater than or equal to 19% of RATED THERMAL POWER and the Reactor Cutback function is disabled, and

ACTION STATEMENTS

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 6 may be inserted no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to the inoperable status.

   c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Off" mode except during CEA group 6 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 6 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 7 inches (indicated position) of all other CEAs in its group.

ACTION 7 - With three or more auto restarts 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 the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

3/4.2 POWER DISTRIBUTION LIMITS

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), provides adequate monitoring of the core power distribution and is 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 limit of 13.4 kW/ft is 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 limit includes appropriate uncertainty and penalty factors necessary to provide a 95/95 confidence level that the maximum linear heat rate calculated by COLSS is greater than or equal to that existing in the core. To ensure that the design margin to safety is maintained, the COLSS computer program includes an F_r measurement uncertainty factor of 1.080, an engineering uncertainty factor of 1.03, a THERMAL POWER measurement uncertainty factor of 1.02 and appropriate uncertainty and penalty factors for flux peaking augmentation and rod bow.

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 limits of Figure 3.2-3 can be maintained by utilizing a predetermined local power density margin and a total core power limit in the CPC trip channels. The above listed uncertainty and penalty factors are also included in the CPCs.

These penalty factors are determined from uncertainties associated with planar radial peaking measurements, engineering heat flux uncertainty, axial densification, software algorithm modelling, computer processing, rod bow, and core power measurement.

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.

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), provides adequate monitoring of the core power distribution and is 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 the minimum DNBR limit includes appropriate penalty factors which provide a 95/95 probability/confidence level that the core power 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 measurements, 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 Figure 3.2-3 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 plus those associated with startup test acceptance criteria 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 inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings.

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ATTACHMENT B

3/4.2 POWER DISTRIBUTION LIMITS

3/4 2.1 LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate limit (of Figure 3.2.1) shall be maintained by one of the following methods, as applicable:

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

b. Operating within the region of acceptable operation of Figure 3.2.1a using any operable CPC channel (when COLSS is out of service and either one or both CEACs is operable).

c. Automatically by CPC (when COLSS is out of service and neither CEAC is operable).

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

ACTION:

With the linear heat rate limit not being maintained as indicated by:

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

2. When COLSS is out of service, operation outside the region of acceptable operation in Figure 3.2.1a;

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. Be in at least HOT STANDBY 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 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 linear heat rate, as indicated on any OPERABLE Local Power Density channels, is within the limits shown on Figure 3.2-1.

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 kW/ft.

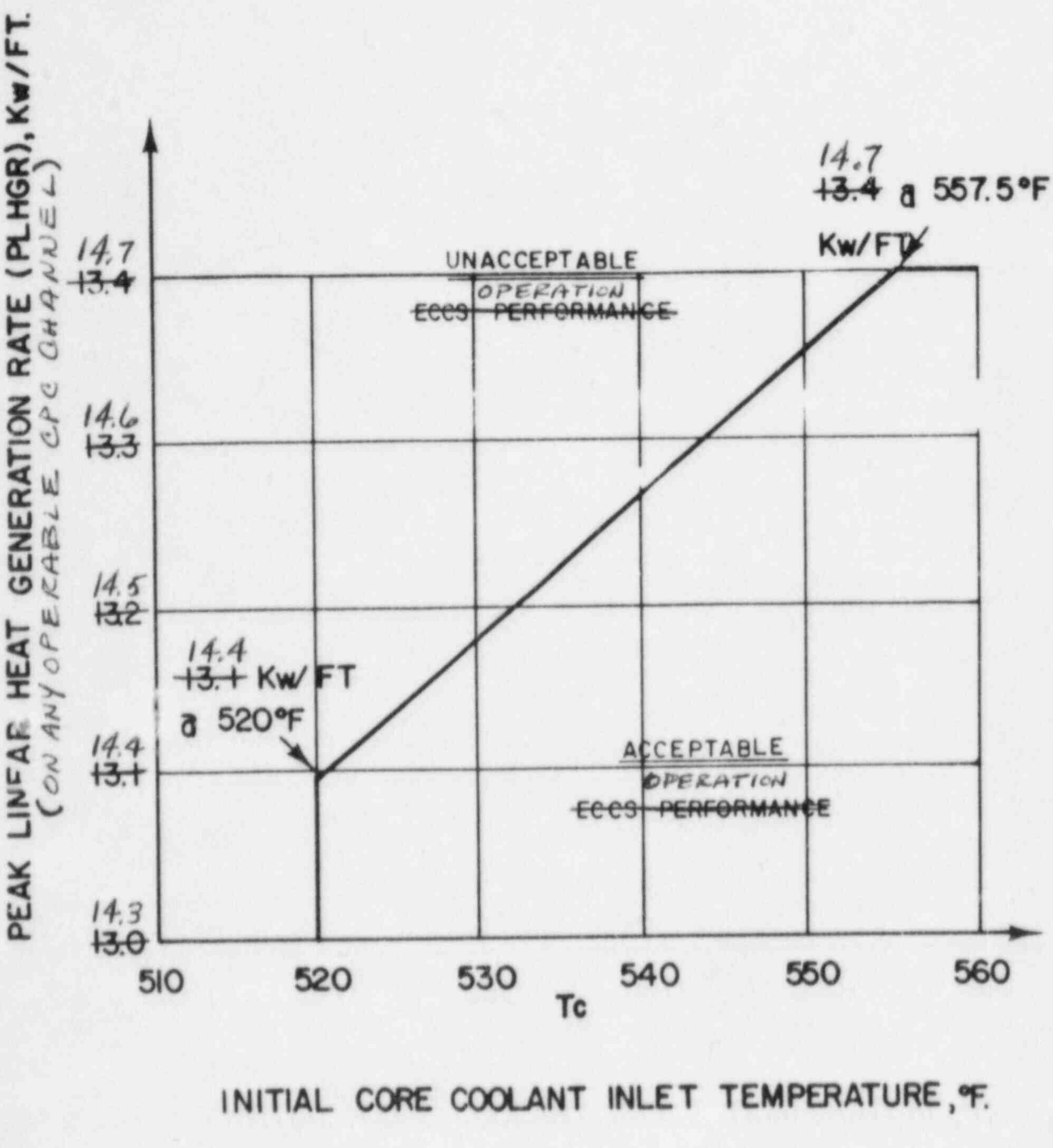

FIGURE 3.2-1a

ALLOWABLE PEAK LINEAR HEAT RATE VS Tc
FOR COLSS OUT OF SERVICE

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 19% 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 using any operable CPC channel (when COLSS is out of service and neither CEACs is operable).

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

ACTION:

With the DNBR margin not being maintained, as indicated by:

1. 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, as applicable;

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

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

b. Be in at least HOT STANDBY 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 channels, is within the limit shown on Figure 3.2-3.

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.

DELETE

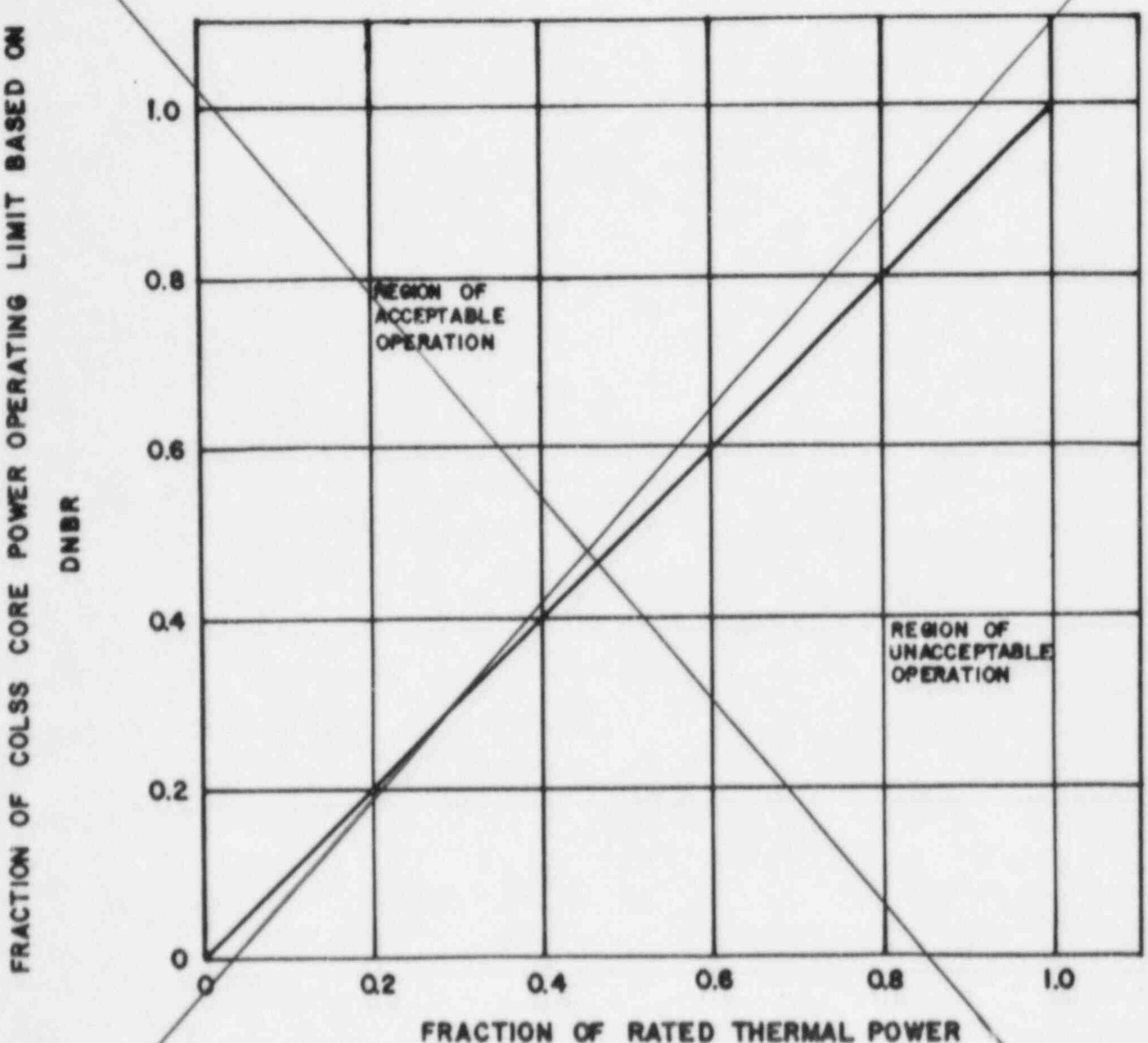

FIGURE 3.2-2

DNBR MARGIN OPERATING LIMIT BASED ON COLSS

COLSS OUT OF SERVICE DNBR LIMIT LINE

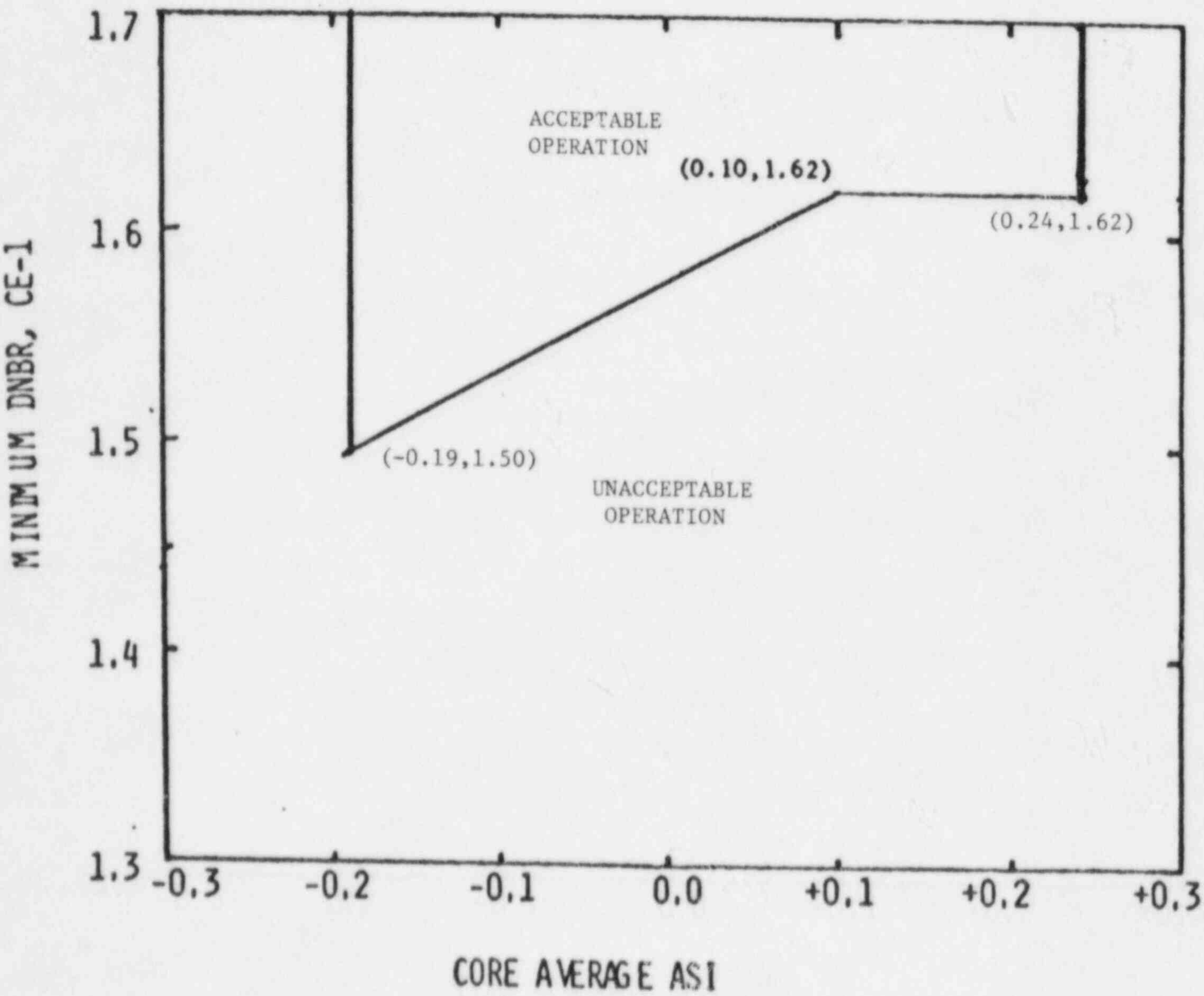

FIGURE 3.2-~~3~~ 2

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

COLSS OUT OF SERVICE DNBR LIMIT LINE

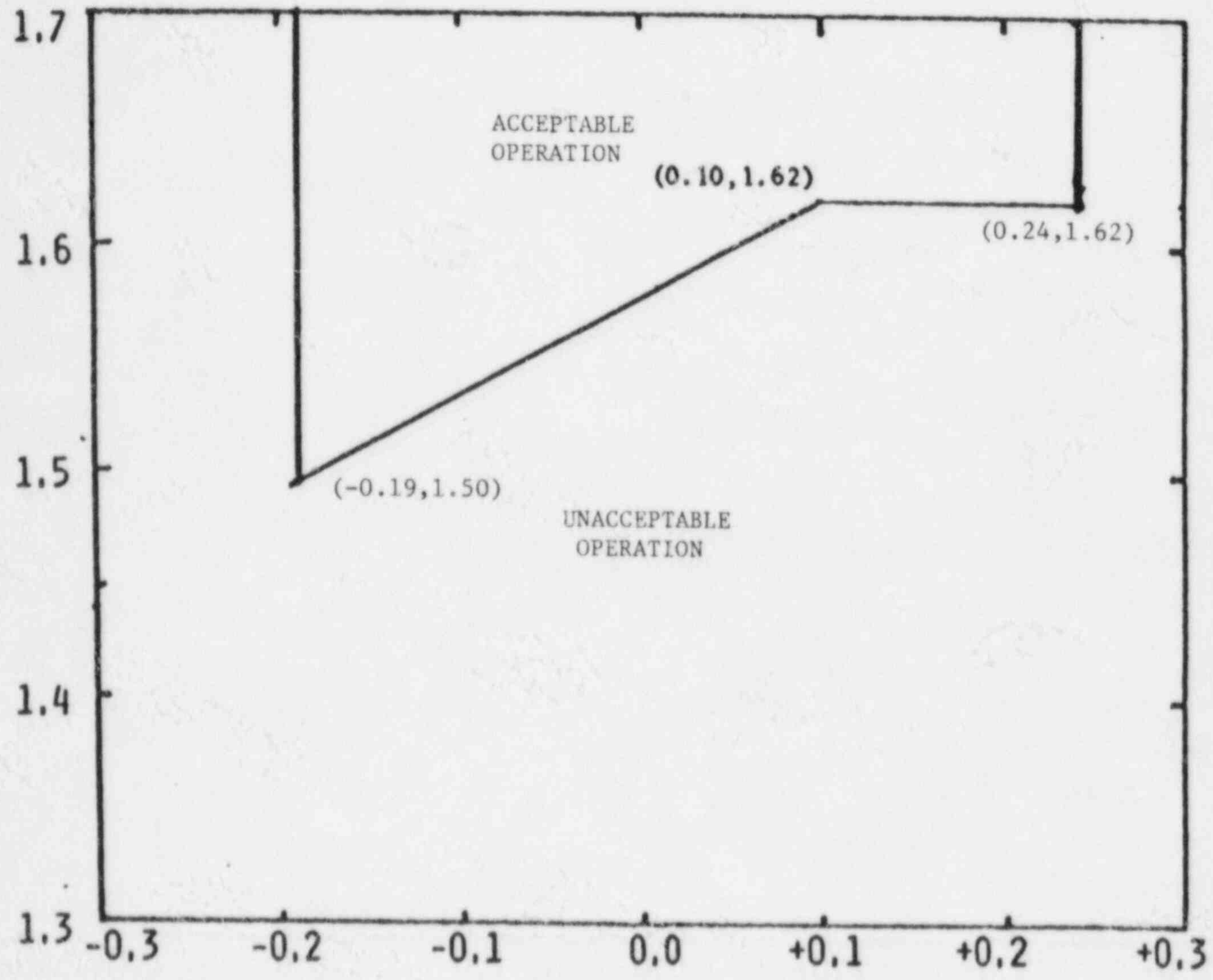

FIGURE 3.2-3

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

3/4.2.7 AXIAL SHAPE INDEX

LIMITING CONDITION FOR OPERATION

3.2.7 The AXIAL SHAPE INDEX (ASI) shall be maintained within the following limits:

a. COLSS OPERABLE
$-0.23 \leq \text{ASI} \leq +0.50$

b. COLSS OUT OF SERVICE (CPC)
$-0.19 \leq \text{ASI} \leq +0.24$

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

ACTION:

With the AXIAL SHAPE INDEX outside its above limits, restore the AXIAL SHAPE INDEX to within its limit within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.7 The AXIAL SHAPE INDEX shall be determined to be within its limit at least once per 12 hours using the COLSS or any OPERABLE Core Protection Calculator channel.

*See Special Test Exception 3.10.2.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

2.	Pressurizer Pressure - High	Pressurizer Pressure - High Local Power Density - High DNBR - Low
3.	Containment Pressure - (RPS) High	Containment Pressure - High Contrinment Pressure - High (ESF)
4.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS 1 and 2)
5.	Steam Generator Level	Steam Generator Level - Low Steam Generator Level - High Steam Generator ΔP (EFAS)
6.	Core Protection Calculator	Local Power Density - High DNBR - Low

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 those required by the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the reactor trip breakers of the inoperable channel are placed in the tripped condition within 1 hour; otherwise, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed 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 7 inches (indicated position) of all other CEAs in its group.

b. With both CEACs inoperable, operation may continue provided that:

   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

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), provides adequate monitoring of the core power distribution and is 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 limit of Figure 3.2-1 is 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 limit includes appropriate uncertainty and penalty factors necessary to provide a 95/95 confidence level that the maximum linear heat rate calculated by COLSS is greater than or equal to that existing in the core. To ensure that the design margin to safety is maintained, the COLSS computer program includes an F_{xy} measurement uncertainty factor of 1.053, an engineering uncertainty factor of 1.03, a THERMAL POWER measurement uncertainty factor of 1.02 and appropriate uncertainty and penalty factors for flux peaking augmentation and rod bow.

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 less accurate 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 limits of Figure 3.2-1a can be maintained by utilizing a predetermined local power density margin and a total core power limit in the CPC trip channels. The above listed uncertainty and penalty factors are also included in the CPCs.

These penalty factors are determined from uncertainties associated with planar radial peaking measurements, engineering heat flux uncertainty, axial densification, software algorithm modelling, computer processing, rod bow, and core power measurement.

The additional uncertainty terms included in the CPC's for transient protection are credited in Figure 3.2-1a since this curve is intended to monitor the LCO only during steady state operation.

In addition, when COLSS is out of service and both CEAC's are inoperable, the 57% penalty applied automatically in CPC can be credited in the CPC linear heat rate calculation since it is required only for transient protection. In this case, Figure 3.2-1 is automatically maintained by the CPC trip limit.

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.

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), provides adequate monitoring of the core power distribution and is 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. The COLSS calculation of core power operating limit based on the minimum DNBR limit includes appropriate penalty factors which provide a 95/95 probability/confidence level that the core power 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 measurements, 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 Figure 3.2-3 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 plus those associated with startup test acceptance criteria are also included in the CPCs which assume a minimum ocre power of 20% of RATED THERMAL POWER. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being less accurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings.

NPF-38-10

DESCRIPTION AND SAFETY ANALYSIS

OF PROPOSED CHANGE NPF-38-10

This is a request to revise Administrative Control 6.4.1 to correct a typographical error.

Existing Specification

See attachment "A"

Proposed Specification

See Attachment "B"

Description

Administrative Control 6.4.1 describes the requirements for the retraining and replacement training program at Waterford 3, including reference to ANSI 3.1-1978, "For Selection and Training of Nuclear Power Plant Personnel".

To meet the intent of Administrative Control 6.4.1, the correct citation in ANSI 3.1-1978 is Section 5.5 entitled "Operator Retraining and Replacement Training". However, due to a typographical error, Administrative Control 6.4.1 presently incorrectly cites Section 5.2 of ANSI 3.1-1978 entitled "Training of Personnel to Be Licensed by the NRC". The proposed change corrects this error by referencing Section 5.5 of ANSI 3.1-1978.

Safety Analysis

The proposed change described above shall be deemed to involve a significant hazards consideration if there is a positive finding in any of the following areas:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of any accident previously evaluated?

   Response: NO

   This change is solely for the purpose of correcting a typographical error and has no effect on plant operations. Therefore, the proposed change will not involve an increase in the probability or consequences of any accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: NO

This change is solely for the purpose of correcting a typographical error and has no effect on plant operations. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

Response: NO

This change is solely for the purpose of correcting a typographical error and has no effect on plant operations. Therefore, the proposed change will not involve a reduction in a margin of safety.

The Commission has provided guidance concerning the application of standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards considerations. Example (i) relates to 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.

In this case, the proposed change is similar to Example (i) in that correction of a typographical error is requested.

Safety and Significant Hazards Determination

Based on the above Safety Analysis, it is concluded that: (1) the proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.91; and (2) there is a reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

NPF-38-10

ATTACHMENT A

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978 except that:

a. The Radiation Protection Superintendent shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, September 1975.

b. Personnel in the Health Physics, Chemistry and Radwaste Departments shall meet or exceed the minimum qualifications of ANSI N18.1-1971.

c. The licensed Operators and Senior Operators shall also meet or exceed the minimum qualifications of the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees.

d. Personnel in the Plant Quality Department, and other staff personnel who peform inspection, examination, and testing functions, shall meet or exceed the minimum qualifications of Regulatory Guide 1.58, Rev. 1, September 1980. (Endorses ANSI N45.2.6-1978)

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Training Manager-Nuclear and shall meet or exceed the requirements and recommendations of Section 5.2 of ANSI 3.1-1978 and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

6.5 REVIEW AND AUDIT

6.5.1 PLANT OPERATIONS REVIEW COMMITTEE (PORC)

FUNCTION

6.5.1.1 The PORC shall function to advise the Plant Manager-Nuclear on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PORC shall be composed of the:

Chairman:	Assistant Plant Manager-Nuclear (Plant Technical Services or Operations and Maintenance)
Vice Chairman:	Technical Support Superintendent-Nuclear
Member:	Maintenance Superintendent-Nuclear
Member:	Operations Superintendent-Nuclear
Member:	Radiation Protection Superintendent-Nuclear
Member:	Quality Control Manager-Nuclear

NPF-38-10

ATTACHMENT B

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978 except that:

a. The Radiation Protection Superintendent shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, September 1975.

b. Personnel in the Health Physics, Chemistry and Radwaste Departments shall meet or exceed the minimum qualifications of ANSI N18.1-1971.

c. The licensed Operators and Senior Operators shall also meet or exceed the minimum qualifications of the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees.

d. Personnel in the Plant Quality Department, and other staff personnel who peform inspection, examination, and testing functions, shall meet or exceed the minimum qualifications of Regulatory Guide 1.58, Rev. 1, September 1980. (Endorses ANSI N45.2.6-1978)

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Training Manager-Nuclear and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI 3.1-1978 and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

6.5 REVIEW AND AUDIT

6.5.1 PLANT OPERATIONS REVIEW COMMITTEE (PORC)

FUNCTION

6.5.1.1 The PORC shall function to advise the Plant Manager-Nuclear on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PORC shall be composed of the:

Chairman:	Assistant Plant Manager-Nuclear (Plant Technical Services or Operations and Maintenance)
Vice Chairman:	Technical Support Superintendent-Nuclear
Member:	Maintenance Superintendent-Nuclear
Member:	Operations Superintendent-Nuclear
Member:	Radiation Protection Superintendent-Nuclear
Member:	Quality Control Manager-Nuclear

NPF-38-11

DESCRIPTION AND SAFETY ANALYSIS

OF PROPOSED CHANGE NPF-38-11

This is a request to revise the applicability of Technical Specification 3.4.8.3, Overpressure Protection Systems.

Existing Specification

See Attachment A

Proposed Specification

See Attachment B

Description

Technical Specification 3.4.8.3 defines the requirements for low temperature overpressure protection (LTOP) provided by the shutdown cooling system including the applicable operation modes. The proposed change revises the applicability in Mode 4 to allow a lower RCS temperature during inservice leak and hydrostatic testing without LTOP in effect. This change is necessary to allow compliance with the requirements of Technical Specification 3.4.9 which requires that the integrity of all ASME Code Class 1, 2 and 3 components be maintained. (Note: LTOP is required at all times, either through the shutdown cooling system relief valves or the primary safety valves. When this document indicates that LTOP is not required, the reference is to the LTOP provided by the shutdown cooling system only.)

In the event that a Code Class 1 component does not meet the integrity requirements of Technical Specification 3.4.9 (e.g. a weld repair) Action Statement (a) requires that integrity be restored prior to increasing the RCS temperature more than 70°F above the minimum temperature required by NDT considerations - presently 202°F (Lowest Service Temperature of Figures 3.4-2 and 3.4-3). Restoring integrity includes a hydrostatic test per the ASME Code at approximately 2400 psia which, by Action Statement (a), must be done prior to increasing the RCS temperature above 272°F.

The present restriction in Technical Specification 3.4.8.3 requiring that LTOP be implemented below 285°F (i.e. Shutdown Cooling System Relief Valves aligned and lift setting less than or equal to 430 psia) will not allow the hydrostatic test to be performed. Additionally, Action Statement (b) of Technical Specification 3.4.8.3 does not allow sufficient time to voluntarily enter the action, and complete the necessary preparations, pressurization, inspections and depressurization per ASME Code requirements. By lowering the Mode 4 temperature to 260°F (for inservice leak and hydrostatic testing only) at which Technical Specification 3.4.8.3 becomes applicable, compliance with Action Statement (a) of Technical Specification 3.4.9 will be allowed with no reduction in safety margin.

Safety Analysis

The proposed change described above shall be deemed to involve a significant hazards consideration if there is a positive finding in any of the following areas:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of any accident previously evaluated?

   Response: NO

   The present temperature of 285°F specified in Technical Specification 3.4.8.3 was established based on the most restrictive heatup or cooldown induced stress condition (allowed by Technical Specification 3.4.8.1) which is necessary to ensure compliance with 10 CFR 50 Appendix G. The limiting case is a 50°F/HR heatup which requires pressure below the limit shown on Figure 3.4-2. At 285°F, the allowable pressure limit is 2500 psia, corresponding to the lift pressure of the primary safety valve required by Technical Specification 3.4.2.1. Above this temperature the safety valve provides overpressure protection and LTOP is no longer required. All other allowed heatup and cooldown rates are bracketed by this condition and are conservative.

   For the proposed change, restricting temperature changes during inservice hydrostatic and leak testing operations to less than or equal to 10°F in any one hour period in Technical Specification 3.4.8.1g reduces thermal stresses and permits RCS pressure of 2500 psia at or above 260°F. Under this condition the LTOP provided by Technical Specification 3.4.8.3 is not required to comply with Appendix G (refer to Figure 3.4-2, Inservice Test curve) and no increased probability of brittle fracture of RCS components results. Therefore, the proposed change will not involve an increase in the probability or consequences of any accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: NO

   The only consideration involved in determination of the appropriate LTOP temperature is that of brittle fracture. It is necessary to restrict the combined stress conditions in the RCS materials due to thermal differentials and pressure induced stresses to an acceptable level. This is accomplished through compliance with Technical Specification 3.4.8.1. LTOP protection is provided to limit the maximum RCS pressure from certain postulated transients whenever the RCS safety valve setpoint is not sufficient to limit the maximum pressure. Restricting the allowable temperature change during inservice hydrostatic and leak testing results in lower thermal stresses, compensating for the higher allowable pressure up to the setting of the RCS Safety Valve. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

   Response: NO

   Figure 3.4-2 of Technical Specification 3.4.8.1 defines the safety margin required by 10 CFR 50 Appendix G. The proposed change does not affect compliance with Technical Specification 3.4.8.1. Therefore, the proposed change will not involve a reduction in a margin of safety.

The Commission has provided guidance concerning the application of standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards considerations. Example (vi) relates to a change which may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria specified in the Standard Review Plan.

In this case, the proposed change described above results in the continued compliance with the criteria of 10 CFR 50 Apepndix G and the Standard Review Plan Section 5.2.2 including Branch Technical Position RSB 5-2. The proposed change does not affect compliance with the fracture toughness requirements of Appendix G due to the restriction on heatup and cooldown rates imposed by Technical Specification 3.4.8.1(g) and, therefore, safety margins are not reduced.

Safety and Significant Hazards Determination

Based on the above Safety Analysis, it is concluded that: (1) the proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.91; and (2) there is a reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

NPF-38-11

ATTACHMENT A

LIMITING CONDITION FOR OPERATION

3.4.8.3 Low temperature overpressure protection shall be provided by:

a. At least one of the following overpressure protection systems being OPERABLE:

   1. Both OPERABLE Shutdown Cooling (SDC) System suction line relief valves (SI-406A and SI-406B) each with a lift setting of less than or equal to 430 psia aligned to the Reactor Coolant System, or,

   2. The Reactor Coolant System depressurized with an RCS vent of greater than or equal to 5.6 square inches.

b. Establishing less than 100°F ΔT between RCS and steam generator temperature or ensuring the pressurizer water volume is less than 900 cubic feet (62.5%), prior to starting any reactor coolant pump.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 285°F, MODE 5, and MODE 6 with the reactor vessel head on.

ACTION:

a. With one Shutdown Cooling System suction line relief valve inoperable, restore the inoperable valve to OPERABLE status within 7 days, or be in at least COLD SHUTDOWN and depressurize and vent the RCS within the next 8 hours.

b. With no Shutdown Cooling System suction line relief valves OPERABLE and capable of providing Reactor Coolant System overpressure protection, either:

   1. Restore at least one Shutdown Cooling System suction relief valve to OPERABLE status within 1 hour, or

   2. Be in at least COLD SHUTDOWN and depressurize and vent the RCS within the next 8 hours.

c. In the event either the Shutdown Cooling System suction relief valves(s) or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the Shutdown Cooling System suction relief valve(s) or RCS vent(s) on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

NPF-38-11

ATTACHMENT B

LIMITING CONDITION FOR OPERATION

3.4.8.3 Low temperature overpressure protection shall be provided by:

a. At least one of the following overpressure protection systems being OPERABLE:

1. Both OPERABLE Shutdown Cooling (SDC) System suction line relief valves (SI-406A and SI-406B) each with a lift setting of less than or equal to 430 psia aligned to the Reactor Coolant System, or,

2. The Reactor Coolant System depressurized with an RCS vent of greater than or equal to 5.6 square inches.

b. Establishing less than 100°F ΔT between RCS and steam generator temperature or ensuring the pressurizer water volume is less than 900 cubic feet (62.5%), prior to starting any reactor coolant pump.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 285°F#, MODE 5, and MODE 6 with the reactor vessel head on.

ACTION:

a. With one Shutdown Cooling System suction line relief valve inoperable, restore the inoperable valve to OPERABLE status within 7 days, or be in at least COLD SHUTDOWN and depressurize and vent the RCS within the next 8 hours.

b. With no Shutdown Cooling System suction line relief valves OPERABLE and capable of providing Reactor Coolant System overpressure protection, either:

1. Restore at least one Shutdown Cooling System suction relief valve to OPERABLE status within 1 hour, or

2. Be in at least COLD SHUTDOWN and depressurize and vent the RCS within the next 8 hours.

c. In the event either the Shutdown Cooling System suction relief valve(s) or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the Shutdown Cooling System suction relief valve(s) or RCS vent(s) on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

260°F during inservice leak and hydrostatic testing with Reactor Coolant System temperature changes restricted in accordance with Specification 3.4.8.1g.

NPF-38-12

DESCRIPTION AND SAFETY ANALYSIS

OF PROPOSED CHANGE NPF-38-12

This is a request to revise Technical Specification 3.1.1.3, Moderator Temperature Coefficient.

Existing Specification

See Attachment A

Proposed Specification

See Attachment B

Description

Technical Specification 3.1.1.3 defines acceptable values of the moderator temperature coefficient (MTC) for various thermal power levels. Surveillance Requirement 4.1.1.3.2 defines the required MTC measurement frequency, including the requirement that MTC be measured within 7 EFPD of reaching 40 EFPD core burnup.

An MTC measurement will be performed as part of the standard reload startup testing program for Waterford 3. However, core burnup at this stage of the test program is on the order of 5-10 EFPD - less than the minimum 33 EFPD presently required by Surveillance Requirement 4.1.1.3.2. The proposed change would allow the "40 EFPD" MTC measurement to be taken at a thermal power level greater than 15% at any point prior to reaching 40 EFPD core burnup. This change would continue to satisfy the intent of an MTC measurement early in core life while allowing credit for the time-consuming MTC measurement performed during the reload startup testing program.

Safety Analysis

The proposed change described above shall be deemed to involve a significant hazards consideration if there is a positive finding in any of the following areas:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of any accident previously evaluated?

   Response: NO

   Technical Specification 3.1.1.3 exists in order to ensure that the MTC values assumed in the FSAR Safety Analyses are conservative with respect to the measured values. MTC varies slowly as a function of core burnup, being more positive at the beginning of core life and becoming more negative with increasing burnup. Therefore, the earlier in core life that MTC is measured, the closer will be the measured value to the positive limits of

Technical Specification 3.1.1.3. In this sense, the proposed change to allow credit for an earlier MTC measurement is in the conservative direction. Therefore, the proposed change will not involve an increase in the probability or consequences of any accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: NO

   The MTC measurement is performed to confirm that the measured value meets the criteria in the limiting condition for operation of Technical Specification 3.1.1.3. The proposed change does not alter the LCO criteria, rather it allows performance of the MTC measurement at any earlier (more conservative) time than presently allowed. The measured MTC value is not used as an input to any safety-related calculation, setpoint, etc. and thus has no potential for affecting future plant operations. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

   Response: NO

   In performing the MTC measurement during reload startup testing all standard criteria (e.g. xenon equilibrium) will be met to ensure an accurate measurement. With respect to MTC, safety margin is defined by the LCO criteria of Technical Specification 3.1.1.3. Allowing credit for a slightly earlier measurement of MTC has no effect on the existing safety margin. Therefore, the proposed change will not involve a reduction in a margin of safety.

The Commission has provided guidance concerning the application of standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards considerations. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications, (i.e. a more stringent surveillance requirement).

In this case, the proposed change is similar to Example (ii) when LP&L chooses to credit MTC measurements performed earlier than the presently allowed 33 EFPD. MTC measurements taken earlier in core life will result in more positive values - i.e. closer to the limits of Technical Specification 3.1.1.3.

Safety and Significant Hazards Determination

Based on the above Safety Analysis, it is concluded that: (1) the proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.91; and (2) there is a reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

NPF-38-12

ATTACHMENT A

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.2×10^{-4} delta k/k/°F whenever THERMAL POWER is ≤ 70% RATED THERMAL POWER, and

b. Less positive than 0.0×10^{-4} delta k/k/°F whenever THERMAL POWER is > 70% RATED THERMAL POWER, and

c. Less negative than -2.5×10^{-4} delta k/k/°F at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

4.1.1.3.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 EFPD of reaching 40 EFPD core burnup.

c. At any THERMAL POWER, within 7 EFPD of reaching two-thirds of expected core burnup.

*With K_{eff} greater than or equal to 1.0.

#See Special Test Exception 3.10.2.

NPF-38-12

ATTACHMENT B

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.2×10^{-4} delta k/k/°F whenever THERMAL POWER is $\leq$ 70% RATED THERMAL POWER, and

b. Less positive than 0.0×10^{-4} delta k/k/°F whenever THERMAL POWER is > 70% RATED THERMAL POWER, and

c. Less negative than -2.5×10^{-4} delta k/k/°F at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

4.1.1.3.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At greater than 15% of RATED THERMAL POWER, prior to reaching 40 EFPD core burnup.

c. At any THERMAL POWER, within 7 EFPD of reaching two-thirds of expected core burnup.

*With K_{eff} greater than or equal to 1.0.

#See Special Test Exception 3.10.2.