June 3, 1985

Docket No. 50-298

Mr. J. M. Pilant, Technical
Staff Manager
Nuclear Power Group
Nebraska Public Power District
Post Office Box 499
Columbus, Nebraska 68601

Dear Mr. Pilant:

The Commission has issued the enclosed Amendment No. 93 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station. The amendment consists of changes to the Technical Specifications in response to your application dated January 10, 1985, as supplemented by submittal dated February 28, 1985.

The amendment revises the Technical Specifications to support operation of Cooper Nuclear Station during the upcoming fuel Cycle 10 and to expand the flexibility of plant limits to permit operation with barrier-type fuel and hafnium (General Electric Hybrid I) control rods. The Technical Specifications are revised accordingly in the following areas: (1) rod block monitor upscale trip setting, (2) maximum average planar linear heat generation rate curves, (3) minimum critical power ratio curves, and (4) description of control rod materials.

A copy of the Safety Evaluation is also enclosed. Our Safety Evaluation considered the information in your application letters noted above as well as your commitment, by letter dated April 4, 1985, to incorporate the recommendations of General Electric Service Information Letter 380 into plant operating procedures prior to startup for Cycle 10 and to translate the procedures into plant Technical Specifications in a timely manner.

Sincerely,

Original signed by/

Ernest D. Sylvester, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 93 to License No. DPR-46
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

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with corrections

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Mr. J. M. Pilant
Nebraska Public Power District

Cooper Nuclear Station

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 93
License No. DPR-46

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Nebraska Public Power District dated January 10, 1985, as supplemented by submittals dated February 28 and April 4, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the licensee is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Facility Operating License No. DPR-46 is hereby amended to read as follows:

   (2) Technical Specification

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 93, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

for Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 3, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 93

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Revise the Appendix A Technical Specifications as follows. The revised areas are indicated by marginal lines.

Remove	Insert
61	61
62	62
211b	211b
212d	212d
212e	212e
--	212f
--	212g
217	217

TABLE 3.2.C
CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

Function	Trip Level Setting	Minimum Number Of Operable Instrument Channels/Trip System(5)
APRM Upscale (Flow Bias)	$\leq (0.66W + 42\%) \frac{FRP}{MFLPD}$ (2)	2(1)
APRM Upscale (Startup)	$\leq$ 12%	2(1)
APRM Downscale (9)	$\geq$ 2.5%	2(1)
APRM Inoperative	(10b)	2(1)
RBM Upscale (Flow Bias)	$\leq$ 0.66W + (N - 66) (2)	1
RBM Downscale (9)	$\geq$ 2.5%	1
RBM Inoperative	(10c)	1
IRM Upscale (8)	$\leq$ 108/125 of Full Scale	3(1)
IRM Downscale (3)(8)	$\geq$ 2.5%	3(1)
IRM Detector Not Full In (8)		3(1)
IRM Inoperative (8)	(10a)	3(1)
SRM Upscale (8)	$\leq 1 \times 10^5$ Counts/Second	1(1)(6)
SRM Detector Not Full In (4)(8)	($\geq$ 100 cps)	1(1)(6)
SRM Inoperative (8)	(10a)	1(1)(6)
Flow Bias Comparator	$\leq$ 10% Difference In Recirc. Flows	1
Flow Bias Upscale/Inop.	$\leq$ 110% Recirc. Flow	1
SRM Downscale (8)(7)	$\geq$ 3 Counts/Second (11)	1(1)(6)
SDV Water Level High CRD-231E, 234E	$\leq$ 46 inches	1(12)

NOTES FOR TABLE 3.2.C

1. For the startup and run positions of the Reactor Mode Selector Switch, the Control Rod Withdrawal Block Instrumentation trip system shall be operable for each function. The SRM and IRM blocks need not be operable in "Run" mode, and the APRM (flow biased) and RBM rod blocks need not be operable in "Startup" mode. The Control Rod Withdrawal Block Instrumentation trip system is a one out of "n" trip system, and as such requires that only one instrument channel specified in the function column must exceed the Trip Level Setting to cause a rod block. By utilizing the RPS bypass logic (see note 5 below and note 1 of Table 3.1.1) for the Control Rod Withdrawal Block Instrumentation, a sufficient number of instrument channels will always be operable to provide redundant rod withdrawal block protection.

2. W is the recirculation loop flow in percent of design. Trip level setting is in percent of rated power (2381 MWt). N is the RBM setpoint selected (in percent) and is calculated in accordance with the methodology of the latest NRC approved version of NEDE-24011-P-A.

3. IRM downscale is bypassed when it is on its lowest range.

4. This function is bypassed when the count is $\geq$ 100 cps and IRM above range 2.

5. By design one instrument channel; i.e., one APRM or IRM per RPS trip system may be bypassed. For the APRM's and IRM's, the minimum number of channels specified is that minimum number required in each RPS channel and does not refer to a minimum number required by the control rod block instrumentation trip function. By design only one of two RBM's or one of four SRM's may be bypassed. For the SRM's, the minimum number of channels specified is the minimum number required in each of the two circuit loops of the Control Rod Block Instrumentation Trip System. For the RBM's, the minimum number of channels specified is the minimum number required by the Control Rod Block Instrumentation Trip System as a whole (except when a limiting control rod pattern exists and the requirements of Specification 3.3.B.5 apply).

6. IRM channels A,E,C,G all in range 8 or higher bypasses SRM channels A&C functions. IRM channels B,F,D,H all in range 8 or higher bypasses SRM channels B&D functions.

7. This function is bypassed when IRM is above range 2.

8. This function is bypassed when the mode switch is placed in Run.

9. This function is only active when the mode switch is in Run. This function is automatically bypassed when the IRM instrumentation is operable and not high.

10. The inoperative trips are produced by the following functions:

    a. SRM and IRM

        (1) Mode switch not in operate
        (2) Power supply voltage low
        (3) Circuit boards not in circuit

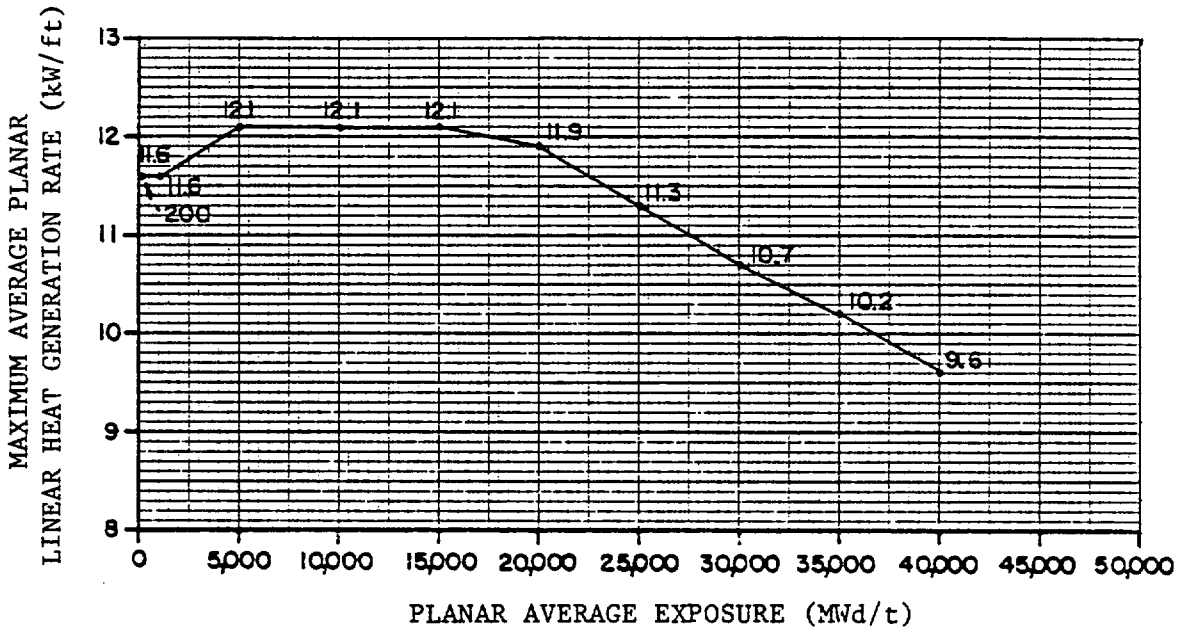

Figure 3.11-1.5 Maximum Average Planar Linear Heat Generation Rate versus Exposure with LPCI Modification and Bypass Flow Holes Plugged, P8DRB265L and BP8DRB265L Fuel.

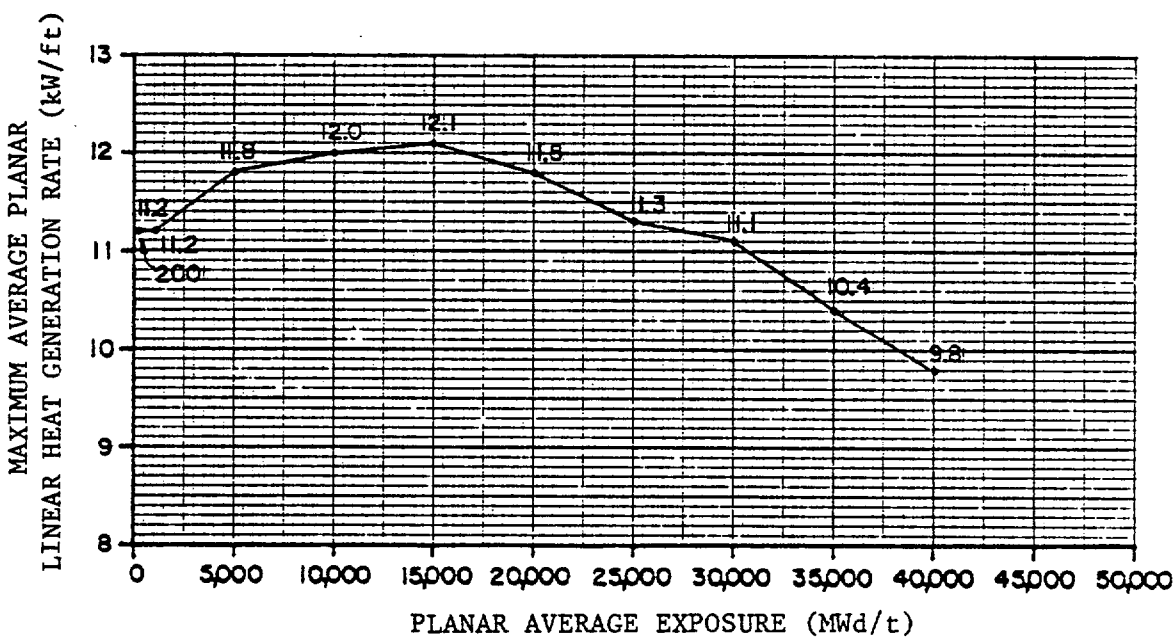

Figure 3.11-1.6 Maximum Average Planar Linear Heat Generation Rate versus Exposure with LPCI Modification and Bypass Flow Holes Plugged, 8DRB283, P8DRB283 and BP8DRB283 Fuel.

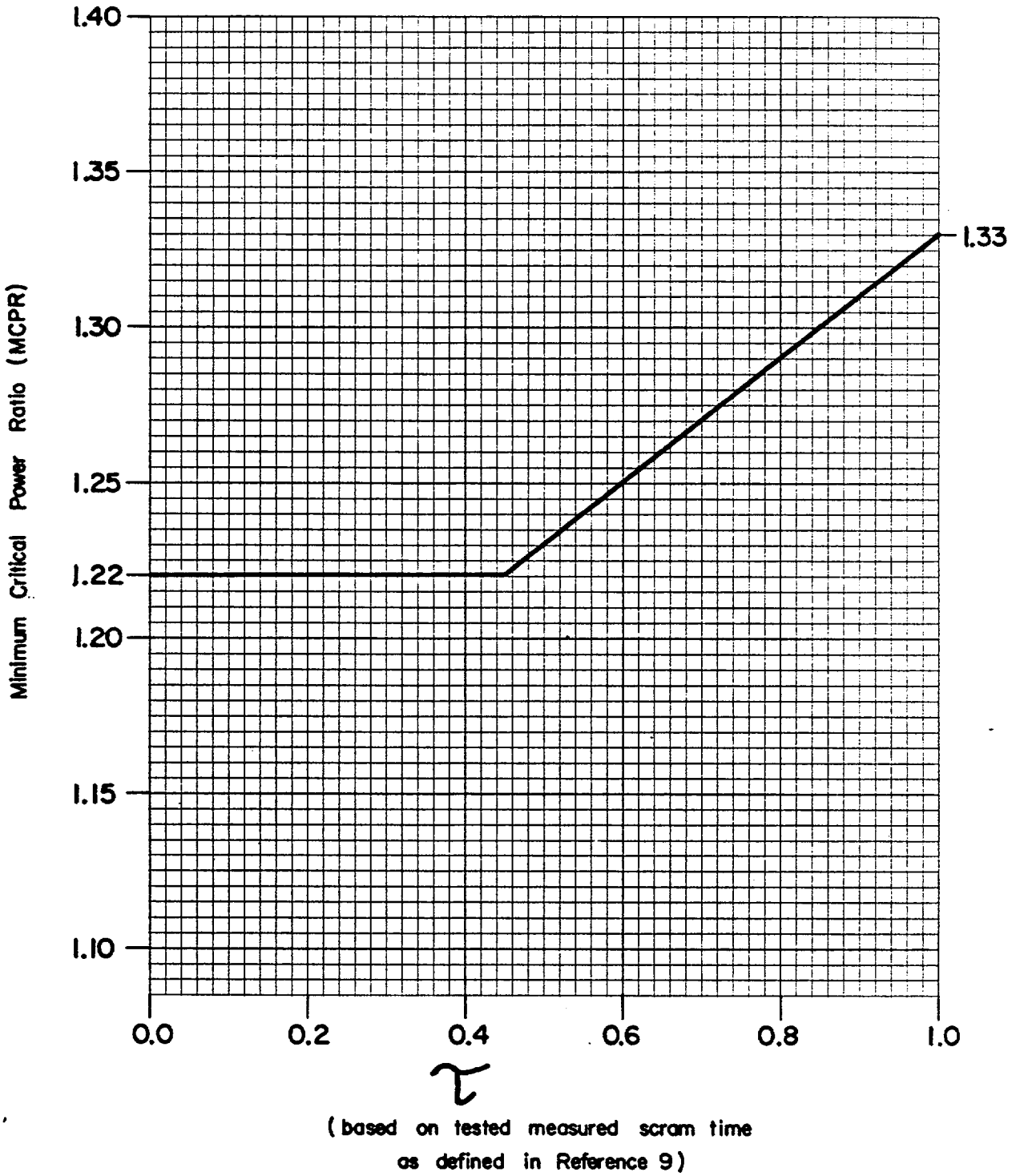

Figure 3.11-2c 8x8R Fuel
(BOC to EOC-1000 MWd/ST)

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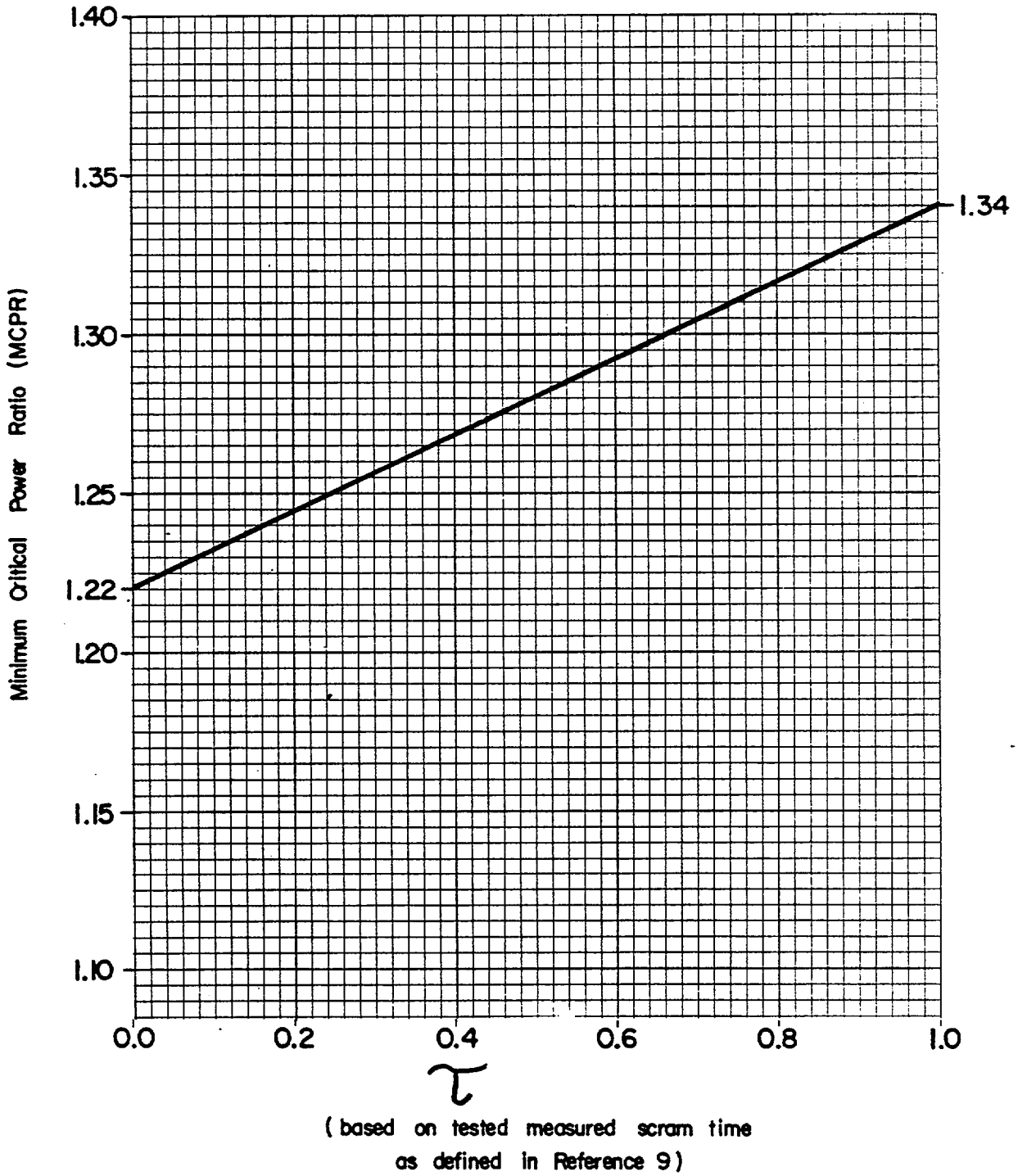
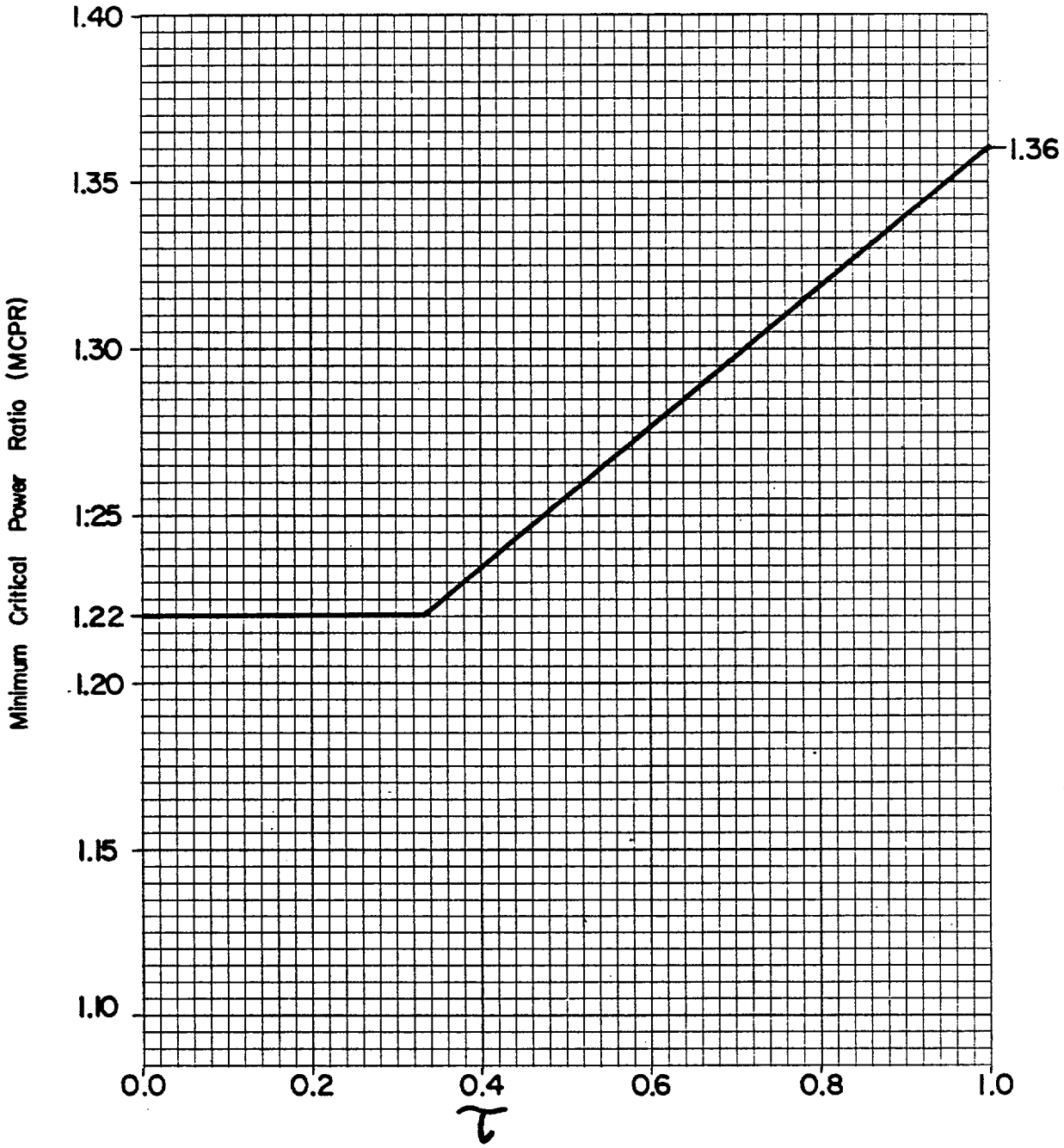


Figure 3.11-2d 8x8R Fuel
(EOC-1000 MWd/ST to EOC)



(based on tested measured scram time as defined in Reference 9)

Figure 3.11-2e P8x8R & BP8 x 8R Fuel (BOC to EOC-1000 MWd/ST)

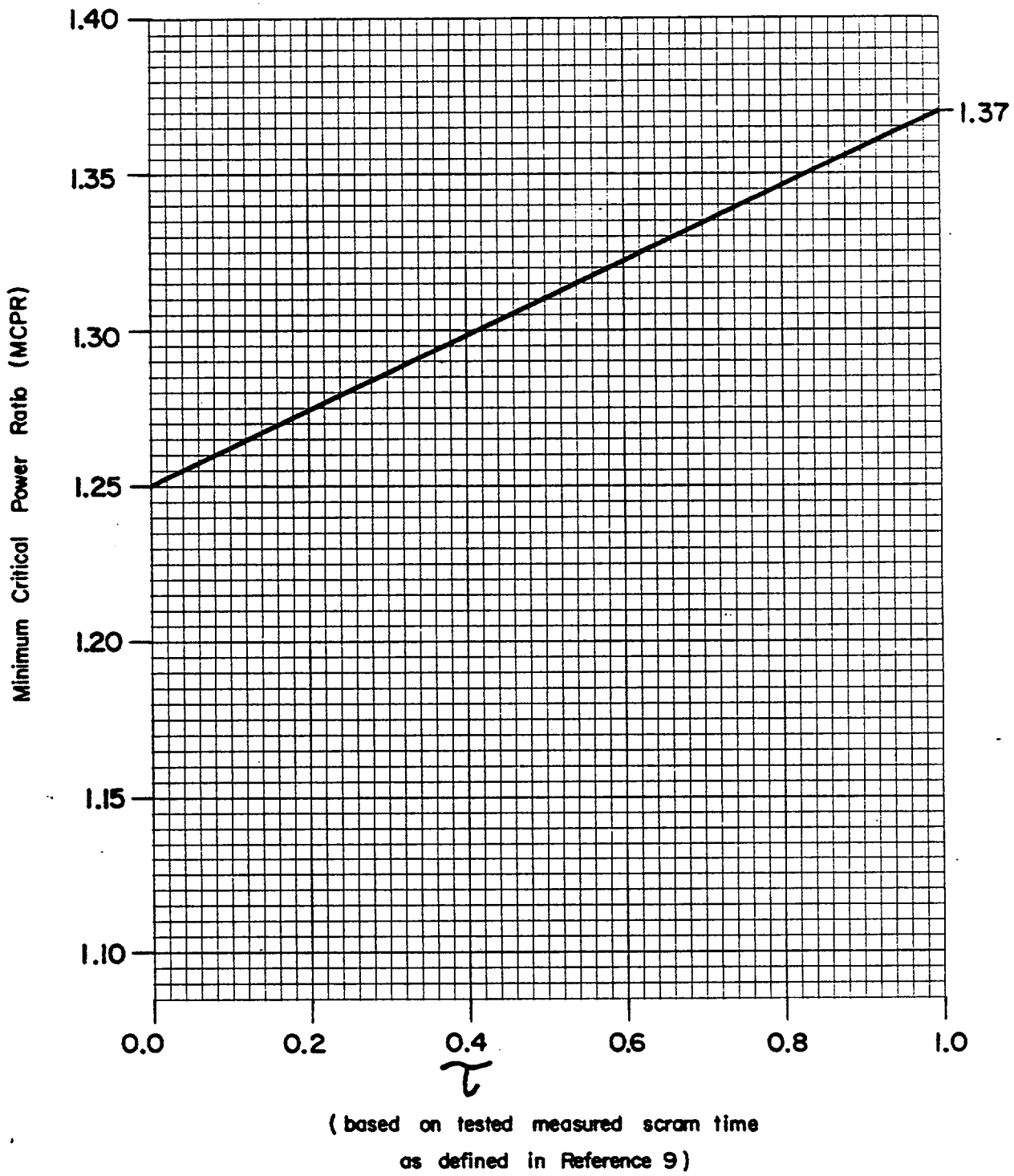


Figure 3.11-2f P8x8R & BP8x8R Fuel
(EOC-1000 MWd/ST to EOC)

5.0 MAJOR DESIGN FEATURES

5.1 Site Features

The Cooper Nuclear Station site is located in Nemaha County, Nebraska, on the west bank of the Missouri River, at river mile 532.5. This part of the river is referred to by the Corps of Engineers as the Lower Brownville Bend. Site coordinates are approximately 40° 21' north latitude and 95° 38' west longitude. The site consists of 1351 acres of land owned by Nebraska Public Power District. About 205 acres of this property is located in Atchison County, Missouri, opposite the Nebraska portion of the station site. The land area upon which the station is constructed is crossed by the Missouri River on the east and is bounded by privately owned property on the north, south, and west. At the west site boundary, a county road and Burlington Northern Railroad spur pass the site.

The reactor (center line) is located approximately 3600 feet from the nearest property boundary. No part of the present property shall be sold or leased by the applicant which would reduce the minimum distance from the reactor to the nearest site boundary to less than 3600 feet without prior NRC approval.

The protected area is formed by a seven foot chain link fence which surrounds the site buildings.

5.2 Reactor

A. The core shall consist of not more than 548 fuel assemblies in any combination of 7x7 (49 fuel rods) and 8x8 (63 fuel rods) and 8x8R/P8x8R (62 fuel rods).

B. The core shall contain 137 cruciform-shaped control rods. The control material shall be boron carbide powder (B_4C) compacted to approximately 70% theoretical density, except for the Hybrid I control rods which contain approximately 15% hafnium.

5.3 Reactor Vessel

The reactor vessel shall be as described in Section IV-20 of the SAR. The applicable design shall be as described in this section of the SAR.

5.4 Containment

A. The principal design parameters for the primary containment shall be as given in Table V-2-1 of the SAR. The applicable design shall be as described in Section XII-2.3 of the SAR.

B. The secondary containment shall be as described in Section V-3.0 of the SAR.

C. Penetrations to the primary containment and piping passing through such

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 93 TO FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By letters dated January 10, 1985 (Ref. 1) and February 28, 1985 (Ref. 8), the Nebraska Public Power District (NPPD) proposed to revise the Technical Specifications for Cooper Nuclear Station (CNS). The proposed amendment would support operation of CNS during the upcoming fuel Cycle 10 (Reload 9) and expand the flexibility of plant limits to permit operation with barrier-type fuel and hafnium (General Electric Hybrid I) control rods. The proposed amendment would revise the following areas: (1) rod block monitor (RBM) upscale trip setting, (2) maximum average planar linear heat generation rate (MAPLHGR) curves, (3) minimum critical power ratio (MCPR) curves, and (4) description of control rod materials. A Supplemental Reload Licensing Submittal, prepared by General Electric and dated November 1984 (Ref. 2), was transmitted by Reference 1 to support the proposed amendment.

The Cooper Nuclear Station Cycle 10 core will consist of 548 fuel assemblies of which 432 are from previous cycles and 116 are new. The new fuel assemblies are of types P8DRB265L and P8DRB283 which are the same types of fuel added in the previous Cycle 9. The core for Cycle 10 consists entirely of these two types of fuel except for 12 type 8DRB283 fuel assemblies which were loaded originally in Cycle 5. Type 8DRB283 fuel assemblies are the same as the type P8DRB283 fuel assemblies except that they are not pressurized.

2.0 EVALUATION

2.1 Fuel Mechanical Design

The 116 new General Electric (GE) fuel assemblies (88 of type P8DRB265L and 28 of type P8DRB283) to be loaded in Cycle 10 are identical to the fuel loaded in previous cycles. These fuel types are standard General Electric reload designs as described in Reference 3. This reference has been approved (Ref. 4) for such use and we conclude that no further review of the fuel design is required. The licensee has also proposed using barrier type fuel (BP8DRB265L and BP8DRB283). This type of fuel has been approved (Ref. 5) and we conclude that no further review of this type of fuel design is required.

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2.2 Nuclear Design

The nuclear design of the Cycle 10 reload core has been performed with methods and techniques that are described in Reference 3. The results of the analyses are given in Reference 2. Those results are within the range of those usually encountered for boiling water reactor (BWR) reloads and are acceptable. In particular the shutdown margin is 0.011 in ΔK. The Standby Liquid Control System is capable of making the system subcritical at cold Xenon-free conditions assuming 600 ppm of boron by an amount of 0.038 ΔK. These results are acceptable. Since they have been obtained by previously approved methods, we conclude that the nuclear design of the Cycle 10 reload core is acceptable.

2.3 Thermal-Hydraulic Design

The thermal-hydraulic (T-H) analysis of the Cycle 10 core was performed with methods and techniques described in Reference 3. Analyses were done at a power level of 2482 thermal megawatts.

2.4 Minimum Critcial Power Ratio

A safety limit value of 1.07 for the core-wide minimum critical power ratio (MCPR) is used for Cycle 10. This value is generic for BWR reloads and is acceptable for Cycle 10.

The operating limit MCPR is obtained by performing analyses of anticipated events in order to determine the reduction in critical power (ΔCPR) resulting from them. Analysis methods, including treatment of uncertainties, are described in Reference 3. The operating limit MCPR is established by adding the largest value of ΔCPR to the safety limit value.

2.5 Thermal-Hydraulic Stability

The Cooper reload submittal relies on the GE cycle specific analysis procedure (GESTAR) to demonstrate that the reactor has sufficient margin to be free of thermal-hydraulic instabilities. The maximum decay ratio calculated in the Cooper submittal is .86. Our evaluation (Ref. 6) of the GE T-H stability methodology has shown that their method has an uncertainty of .2 in calculated decay ratio. Since the Cooper (.86) decay ratio is based on a best estimate calculation their true decay ratio could be as high as 1.06 (.86 + .2). Since a decay ratio greater than 1.00 indicates an undamped oscillation the Cooper analysis does not show any margins from undamped oscillations.

Our evaluation (Ref. 6) of the GE T-H stability methodology also concludes that a core design consisting of approved GE fuel bundles in conjunction with General Electric Service Information Letter (SIL)-380 operating recommendations incorporated into the Technical Specifications is in compliance with GDC 10 and 12 requirements. Since the licensee could not show through analysis that T-H instabilities are prevented by design, he has committed (Ref. 11) to incorporate GE SIL-380 recommendations into plant operating procedures prior

to startup of Cycle 10 and to translate the procedures into plant Technical Specifications in a timely manner.

With the licensee's commitments relative to the GE SIL-380 recommendations (Ref. 11), the staff concludes that thermal hydraulic instability does not pose a safety concern for continued operation of Cooper.

2.6 Transient and Accident Analyses

Transient and accident analysis methods are described in Reference 3. These are the same methods that have been used in previous cycles for CNS and they are acceptable for Cycle 10.

2.7 Pressurization Events

The one-dimensional transient code ODYN has been used to analyze these events. The licensee has elected to use ODYN Option B in which measured rod scram times are used. For this option the pressurization events are not limiting. If Option A scram times are used the Load Rejection Without Bypass event is limiting. Use of the Option B mode is widespread in boiling water reactors and its use is acceptable for CNS.

2.8 Non-Pressurization Events

The licensee has elected to use the generic analysis results for the rod withdrawal event. This has been approved by the staff for BWR reloads and is acceptable for CNS. The rod withdrawal error event is the limiting event for Option B and establishes the operating limit MCPR for Cycle 10 of 1.22. The analysis of this event has been performed by the approved methods of Reference 3 and is acceptable.

2.9 Loss-of-Coolant Accident

The loss-of-coolant accident has been analyzed for Cycle 10 of CNS at a power of 2532 thermal megawatts (2482 x 1.02). The analysis has been performed by the approved methods of Reference 7 and is acceptable.

2.10 Rod Drop Accident

A cycle specific rod drop accident analysis has been performed for Cycle 10 of CNS for both the hot and cold shutdown cases since the parameters of the generic analysis were not bounding for these cases. The result is less than the NRC criterion of 280 calories per gram for the peak enthalpy in both analyses. Since this meets our criterion for this event it is acceptable.

2.11 Technical Specifications

The changes to be made to the Technical Specifications are due to the following circumstances:

1. Page 61 (Table 3.2.C) - The equation for the Rod Block Monitor (RBM) Upscale (Flow Bias) was changed.

2. Page 62 (Notes for Table 3.2.C.) - The note for the RBM in Table 3.2.C was changed. In addition, the staff requested the reference for the method of calculating the value of N, the RBM setpoint selected. This was provided in Reference 8.

3. Page 211b - Figures 3.11-1.5 and 3.11-1.6 - Maximum Average Planar Linear Heat Generation (MAPLHGR) Rate versus Exposure with LPCI Modification and Bypass Flow Holes Plugged. The titles to these figures were changed to include their application to the new barrier fuel.

4. Page 212d - Figure 3.11-2c - Minimum Critical Power Ratio (MCPR) vs τ(based on tested measured scram time as defined in Reference 9) for 8x8R Fuel (BOC to EOC - 1000 MWd/ST). This figure is the result of new analyses.

5. Page 212e - Figure 3.11-2d - MCPR vs τ for 8x8R Fuel (EOC - 1000 MWd/ST to EOC). This figure is the result of new analyses.

6. Page 212f - Figure 3.11-2e - MCPR vs τ for P8x8R and BP8x8R Fuel (BOC to EOC - 1000 MWd/ST). This figure is the result of new analyses.

7. Page 212g - Figure 3.11-2f - MCPR vs τ for P8x8R and BP8x8R Fuel (EOC - 1000 MWd/ST to EOC). This figure is the result of new analyses.

8. Page 217 - Section 5.2.B currently describes the control rod material as boron carbide power (B_4C) compacted to approximately 70 percent theoretical density. In anticipation of the use of the GE Hybrid I Control Rod (HICR) which contains approximately 15 percent hafnium, the licensee wishes to also include this type in the description.

Each of these is discussed below.

2.12 <u>RBM Upscale (Flow Bias) Equation Modification</u>

The trip level setting equation for the RBM Upscale (Flow Bias) was changed to 0.66W + (N-66) from (0.66W+40%). This allows the value of N, the RBM setpoint (%), to be calculated in accordance with the latest NRC approved version of NEDE-24011-A. This change follows approved procedures and is, therefore, acceptable.

2.13 <u>Note to RBM Upscale Equation Modification</u>

This change serves to explain the terms used for the RBM Upscale equation and is acceptable.

2.14 MAPLHGR vs Exposure - Figures 3.11-1.5 and 3.11-1.6

The titles to Figures 3.11-1.5 and 3.11-1.6 were changed to include BP8DRB265L and BP8DRB283 barrier fuel respectively which exhibits the same values as the existing fuel. The term "barrier fuel" stems from the use of a 0.003-inch thick, high purity zirconium liner, i.e., barrier bonded to the inner surface of the Zircaloy-2 portion of the fuel rod cladding. The overall dimensions of the fuel rods are the same as for the GE 8x8 prepressurized retrofit bundle. The use of the barrier fuel was approved in Reference 5 and is, therefore, acceptable.

2.15 MCPR Specification Format - Figure 3.11-2c
8x8R Fuel (BOC to EOC - 1000 MWd/ST)

The cycle Minimum Critical Power Ratio (MCPR) as a function of the parameter τ is presented in curve form. This is from the analyses performed for non-pressurization events and pressurization events (for both the Option B scram time and the Option A [Technical Specification] scram time) in order to establish end points on the curves. The curve of MCPR as a function of τ is consistent with the results of the safety analysis (Ref. 2) and is acceptable. The format of the Technical Specification is similar to that of other plants using Option B and is acceptable.

2.16 MCPR Specification Format - Figure 3.11-2d
8x8R Fuel (EOC-1000 MWd/ST to EOC)

This is acceptable for the same reasons as given for 2.15.

2.17 MCPR Specification Format - Figure 3.11-2e
P8x8R and BP8x8R Fuel (BOC to EOC-1000 MWd/ST)

This is acceptable for the same reasons as given for 2.15.

2.18 MCPR Format - Figure 3.11-2f
P8x8R and BP8x8R Fuel (EOC-1000 MWd/ST to EOC)

This is acceptable for the same reasons as given for 2.15.

2.19 Hybrid I Control Rods (HICR)

The description of the HICR control rods was submitted to NRC by General Electric in Topical Report NEDE-22290-A. The Safety Evaluation of the Type I HICR was reviewed and approved by the staff (Ref. 10). The probability of occurrence or the consequences of an accident would not be increased above those analyzed in the Final Safety Analysis Report (FSAR) because the weight and envelope of the HICR are identical to those of currently used assemblies, and the nuclear and mechanical properties of the HICR do not differ from currently used assemblies in a significant way. The staff has made similar assessments for Monticello, Dresden Unit 2, Hatch Unit 1, and Brunswick 1 and

2. Therefore, the use of the HICR control rods is acceptable and the addition of the HICR description in the Technical Specifications is acceptable.

3.0 SUMMARY

As a result of our review, which is described above, we conclude that the reload and Technical Specification changes proposed by the licensee letters dated January 10 and February 28, 1985, are acceptable. This conclusion is based on the following:

a. Previously approved analysis methods and techniques are employed.

b. The consequences of the transients and accidents which are affected by the reload are acceptable for Cycle 10.

c. The revisions to the Technical Specifications have been found to be acceptable.

d. The licensee has committed (Ref. 11) to incorporate the GE SIL-380 recommendations relative to thermal-hydraulic stability (Section 4.2) into the plant operating procedures prior to startup of Cycle 10 and to translate the procedures into the plant Technical Specifications in a timely manner.

4.0 ENVIRONMENTAL CONSIDERATIONS

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

5.0 CONCLUSIONS

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: H. Balukjian, G. Schwenk, W. Brooks

Dated: June 3, 1985

REFERENCES

1. Letter, L. C. Kuncl, Nebraska Public Power District, to D. B. Vassallo (NRC) dated January 10, 1985.

2. "Supplemental Reload Licensing Submittal for Cooper Nuclear Power Station Unit 1, Reload 9," General Electric Company Report 23A1813 Class 1, November 1984.

3. GESTAR II, General Electric Standard Application for Reactor Fuel, NEDE-24011-A, latest approved version.

4. Approval letter, D. G. Eisenhut (NRC) to R. Gridley (GE) dated May 12, 1978, and Supplements thereto, forming Appendix C to Reference 3.

5. Letter, C. O. Thomas (NRC) to J. S. Charnley (GE) dated April 13, 1983.

6. Letter, C. O. Thomas (NRC) to H. C. Pfefferlen (GE) dated April 24, 1985 - Acceptance for Referencing of Licensing Topical Report NEDE-24011, Rev. 6, Amendment 8, "Thermal Hydraulic Stability Amendment to GESTAR II."

7. "Loss-of-Coolant Accident Analysis Report for Cooper Nuclear Power Station," NEDO-24095, August 1977 (as amended).

8. Letter, J. M. Pilant, Nebraska Public Power District, to D. B. Vassallo (NRC), dated February 28, 1985.

9. Letter, R. H. Buchholz (GE) to P. S. Check (NRC), dated September 5, 1980.

10. Letter, C. O. Thomas (NRC) to J. F. Kapproth (GE), "Acceptance of Referencing of Licensing Topical Report NEDE-22290, Safety Evaluation of the General Electric Hybrid I Control Assembly," dated August 22, 1983.

11. Letter, J. M. Pilant, Nebraska Public Power District, to D. B. Vassallo (NRC), dated April 4, 1985.