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Docket No. 50-298

May 22, 1981

Mr. J. M. Pilant, Director
Licensing & Quality Assurance
Nebraska Public Power District
P. O. Box 499
Columbus, Nebraska 68601

Dear Mr. Pilant:

The Commission has issued the enclosed Amendment No. 70 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 request of March 5, 1981.

The amendment modifies the Technical Specifications to: (1) permit operation of the facility during Cycle 6 with 112 new P8x8R fuel assemblies designed and fabricated by General Electric Company (GE) replacing an equivalent number of 7x7 and 8x8 irradiated fuel assemblies and (2) revise limits based on transient and accident analysis for Cycle 6 core loading which included use of the ODYN code for the analyses of overpressurization transients.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Original Signed by
T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 70 to DPR-46
2. Safety Evaluation
3. Notice

cc w/encls:
See next page

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JUN 02 1981
U.S. NUCLEAR REGULATORY COMMISSION

THIS RELOAD WAS REVIEWED BY SAB & RSB AND NO SER WAS ISSUED

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OFFICE	DL:ORB#2	DL:ORB#2	DL:ORB#2	DL:OR	OELD		
SURNAME	SNorris	BSiegel:ms	TAIppolito	TMNovak	DSWANSON		
DATE	5/18/81	5/18/81	5/18/81	5/18/81	5/19/81		

NRC FORM 318 (10/80) NRCM 0240 OFFICIAL RECORD COPY ☆ USGPO: 1980—329-824

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

May 22, 1981

Docket No. 50-298

Mr. J. M. Pilant, Director
Licensing & Quality Assurance
Nebraska Public Power District
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Thomas A. Ippolito, Chief
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2. Safety Evaluation
3. Notice

cc w/encls:
See next page

Mr. J. M. Pilant
Nebraska Public Power District

cc:

Mr. G. D. Watson, General Counsel
Nebraska Public Power District
P. O. Box 499
Columbus, Nebraska 68601

Mr. Arthur C. Gehr, Attorney
Snell & Wilmer
3100 Valley Center
Phoenix, Arizona 85073

Cooper Nuclear Station
ATTN: Mr. L. Lessor
Station Superintendent
P. O. Box 98
Brownville, Nebraska 68321

Auburn Public Library
118 - 15th Street
Auburn, Nebraska 68305

Mr. Dennis Dubois
USNRC
Resident Inspector
P.O. Box 218
Brownsville, NA 68410

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. 70
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 March 5, 1981 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 license 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 Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 70 , 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

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Dated: May 22, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 70

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Remove the following pages of the Appendix "A" Technical Specifications and replace with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Replace
212	212
-	212b
-	212c
-	212d
-	212e
213	213
214c	214c
214d	214d

LT = Total core length - 12 feet

L = Axial position above bottom of core

G = 18.5 kW/ft for 7x7 fuel bundles
= 13.4 kW/ft for 8x8 fuel bundles

N = 0.038 for 7x7 fuel bundles
= 0.0 for 8x8 fuel bundles

If at any time during steady state operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded action shall then be initiated to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

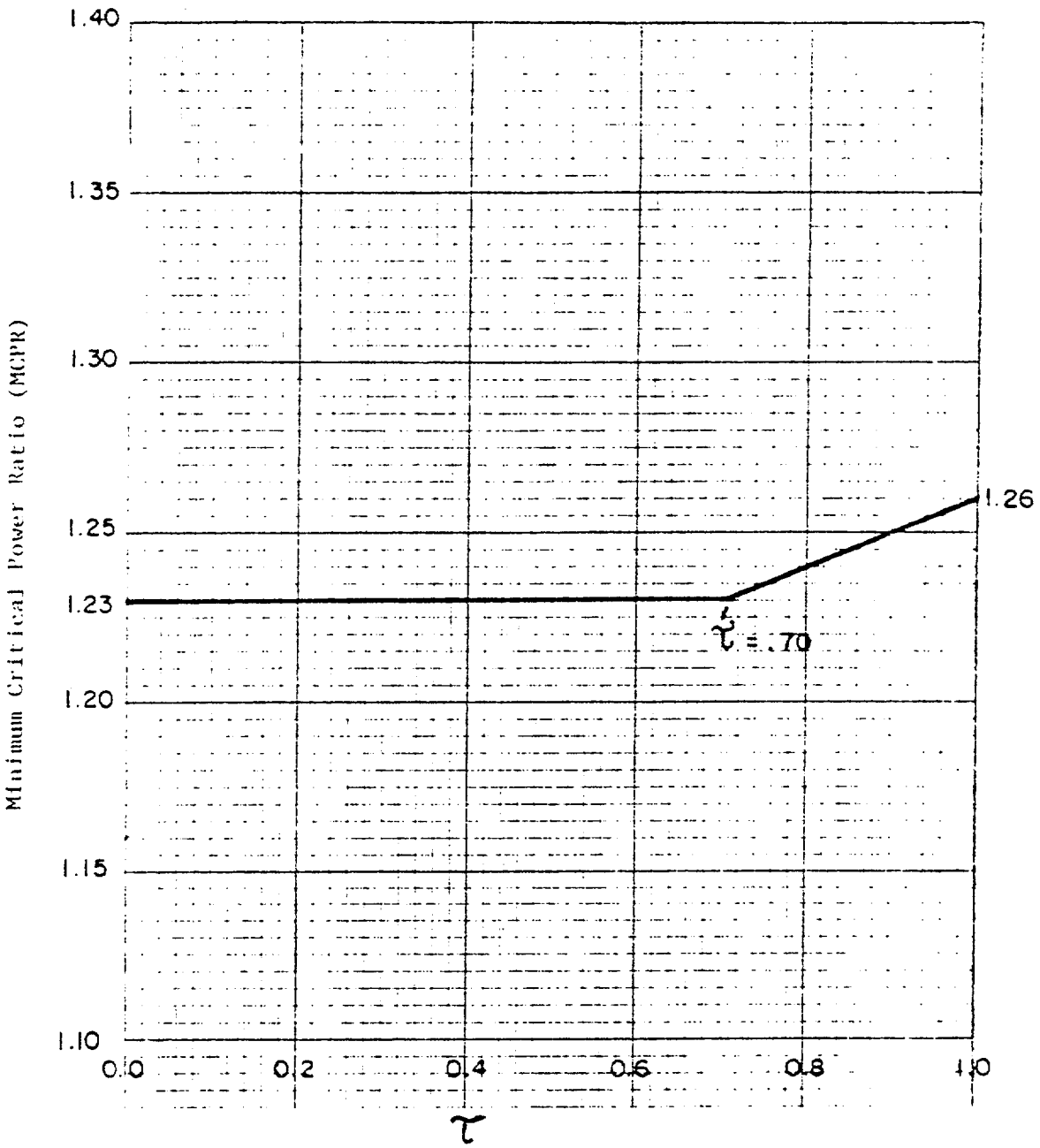
C. Minimum Critical Power Ratio (MCPR)

During steady state power operation the MCPR for each type of fuel at rated power and flow shall not be lower than the limiting value shown in Figure 3.11-2. If, at any time during steady state operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

For core flows other than rated the MCPR shall be the operating limit at rated flow times K_f, where K_f is as shown in Figure 3.11-3.

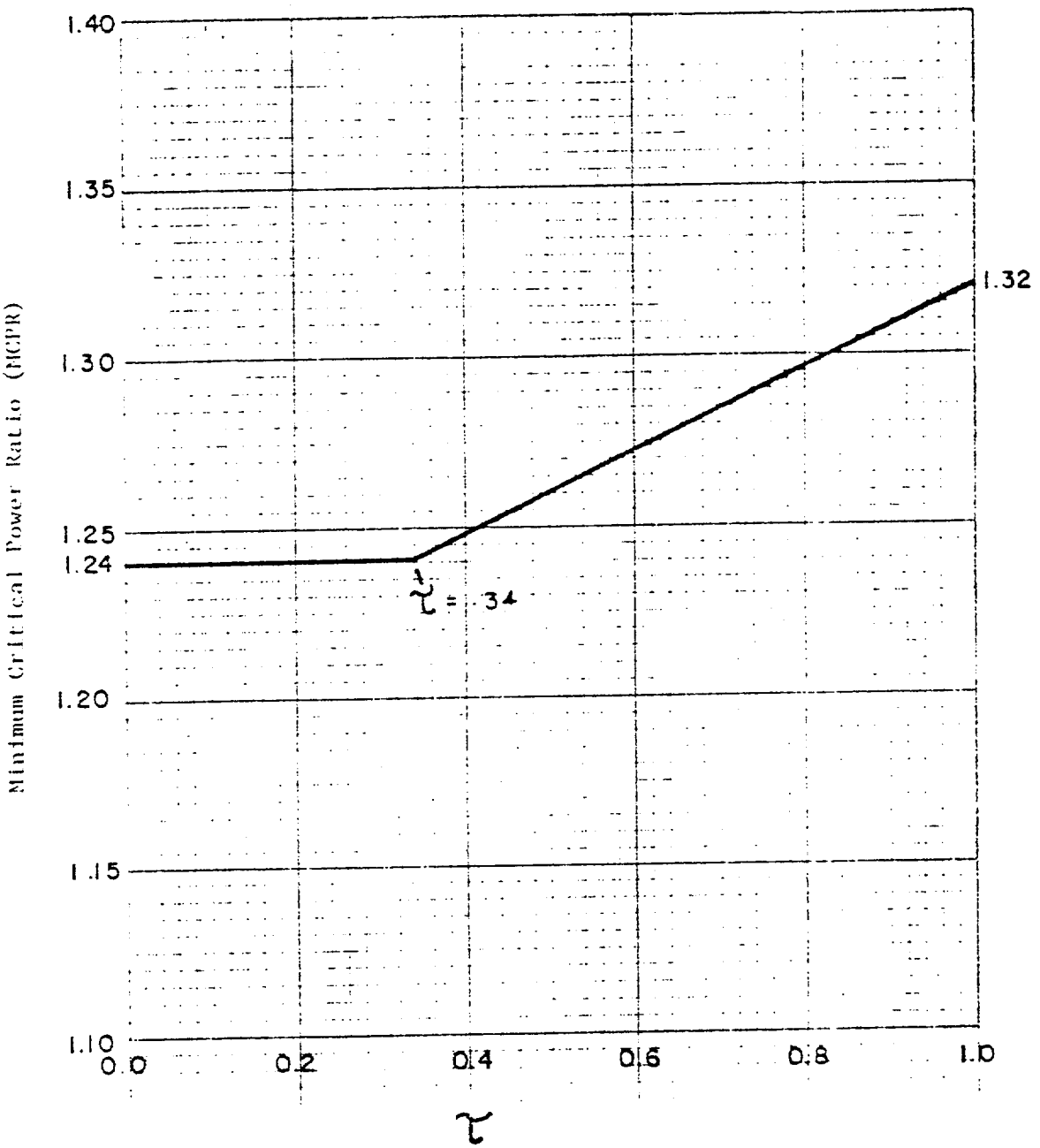
C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at > 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.



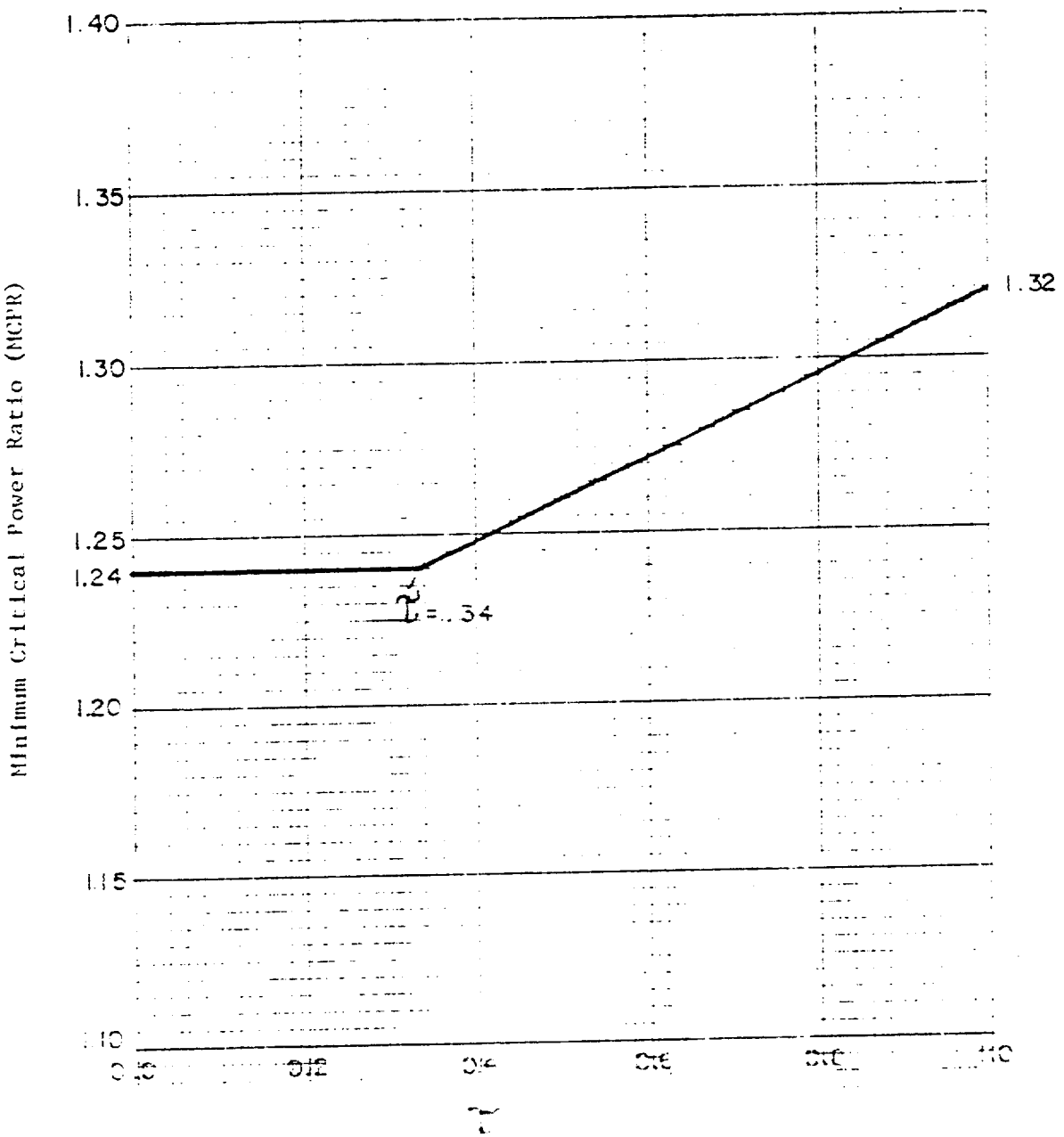
(based on tested measured scram time as defined in Reference 9)

Figure 3.11-2a 7x7 Fuel



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

Figure 3.11-2b 8x8 Fuel


1.40
1.35
1.30
1.25
1.24
1.20
1.15
1.10
1.32
Minimum Critical Power Ratio (MCPR)

[illegible]

[illegible]

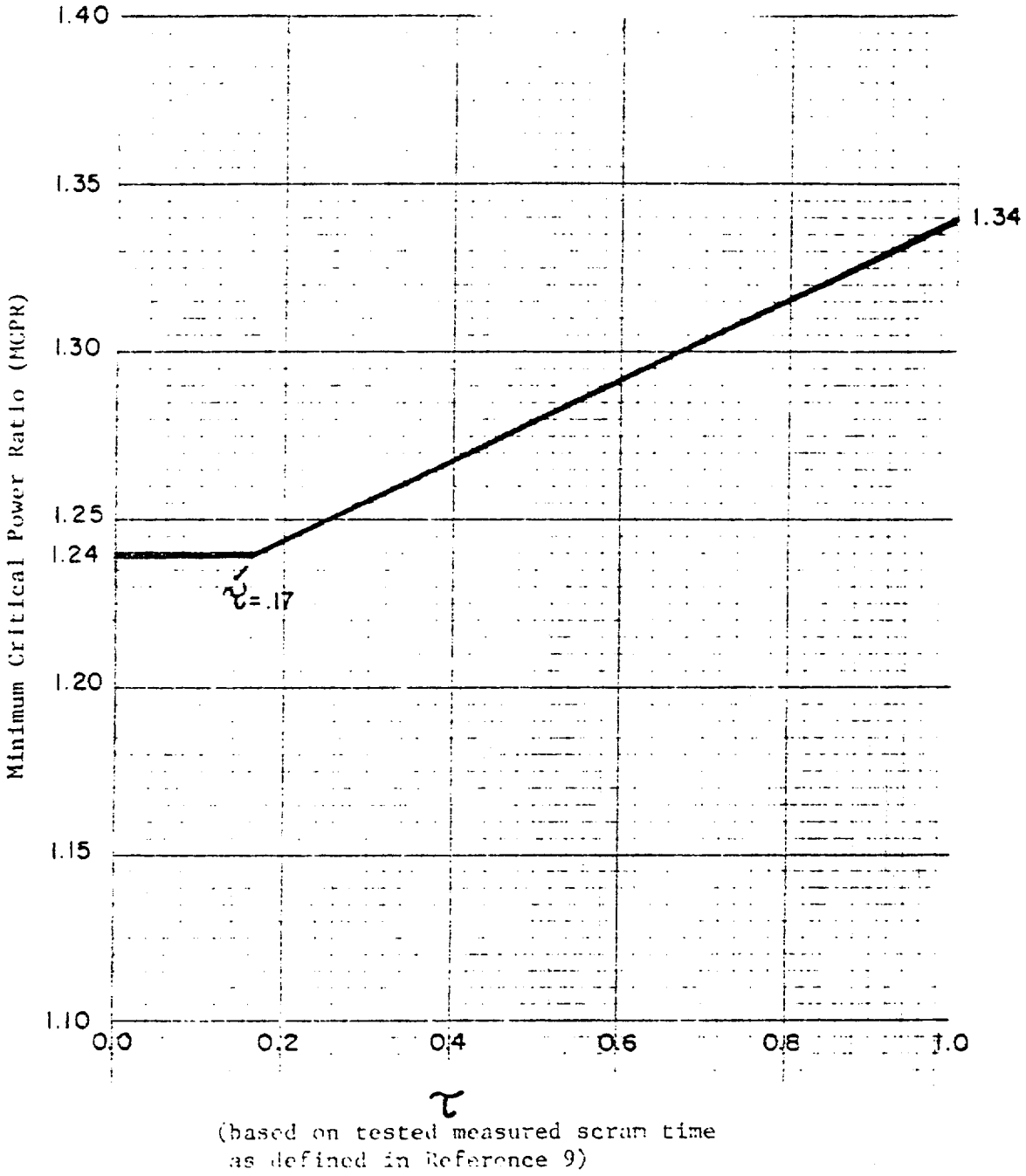
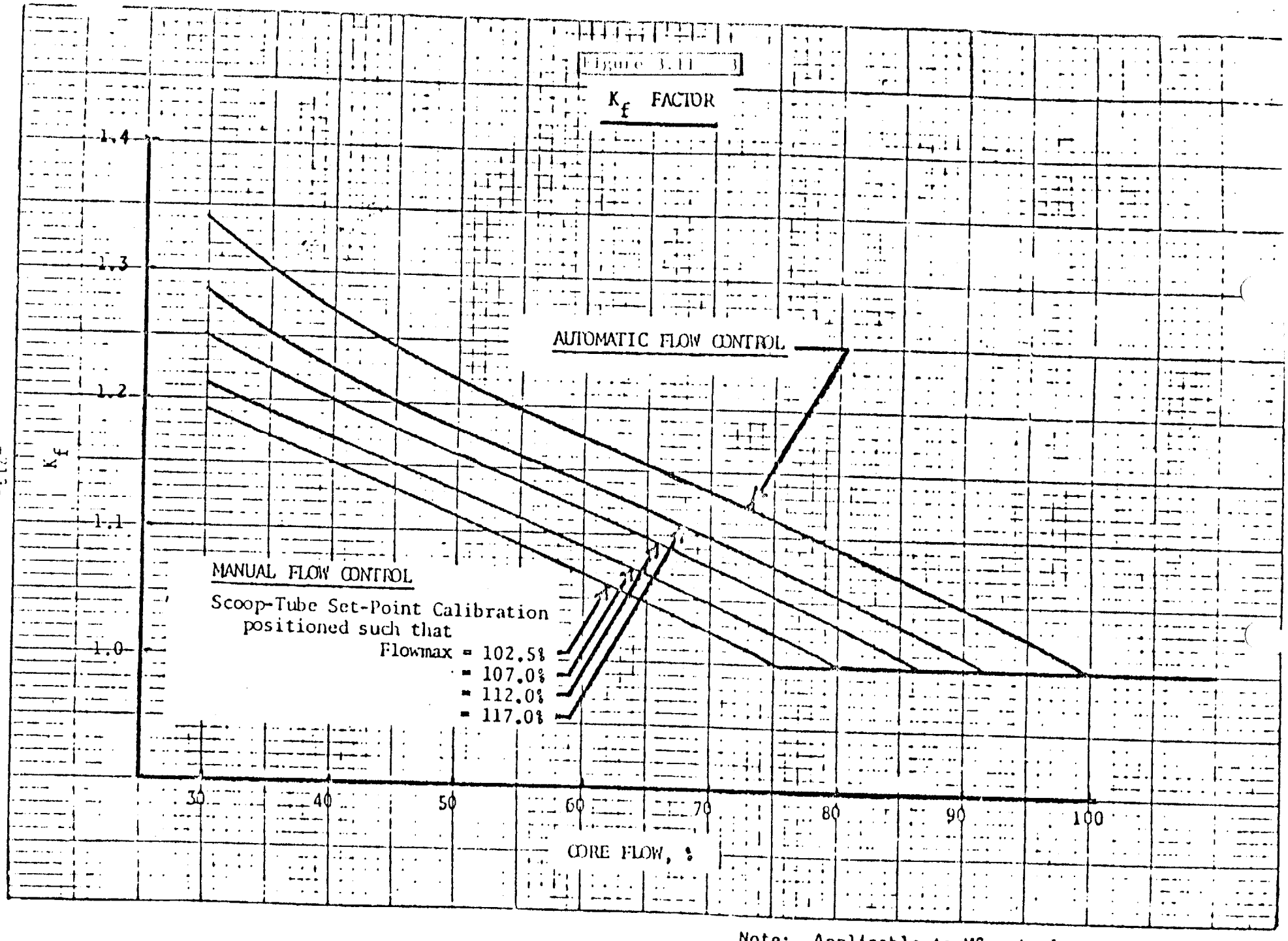


Figure 3.11-2d P8x8R Fuel



Note: Applicable to MG set plants only.

3.11 Bases: (Cont'd)

The limiting transient which determines the required steady state MCPR limit and thus yields the largest ΔCPR is discussed in Reference 5. When added to the safety limit MCPR of 1.07 the deterministic MCPR's are obtained. The required minimum operating limit MCPR's are determined by methods given in References 8 and 9.

Prior to the analysis of abnormal operational transients an initial fuel bundle MCPR was determined. This parameter is based on the bundle flow calculated by a GE multi-channel steady state flow distribution model as described in Section 4 of NEDO-24011[2] and on core parameters shown in Table 5-2 of Reference 2.

The evaluation of a given transient begins with the system initial parameters shown in Table 5-2 of Reference 2 that are input to the GE core dynamic behavior transient computer program described in NEDO-10802[3] and NEDO-24154[7]. The outputs of the program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic SCAT code described in NEDE-20566[4]. The principal result of this evaluation is the reduction in MCPR caused by the transient.

D. MCPR Limits for Core Flows Other than Rated

The purpose of the K_f factor is to define operating limits at other than rated flow conditions. At less than 100% flow, the required MCPR is the product of the operating limit MCPR and the K_f factor. Specifically, the K_f factor provides the required thermal margin to protect against a flow increase transient. The most limiting transient initiated from less than rated flow conditions is the recirculation pump speed up caused by a motor-generator speed control failure.

For operation in the automatic flow control mode, the K_f factors assure that the operating limit MCPR will not be violated should the most limiting transient occur at less than rated flow. In the manual flow control mode, the K_f factors assure that the Safety Limit MCPR will not be violated for the same postulated transient event.

3.11 Bases: (Cont'd)

The K_f factor curves shown in Figure 3.11-3 were developed generically which are applicable to all BWR/2, BWR/3, and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to rated thermal power at rated core flow.

For the manual flow control mode, the K_f factors were calculated such that at the maximum flow state (as limited by the pump scoop tube set point) and the corresponding core power (along the rated flow control line), the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPR's were calculated at different points along the rated flow control line corresponding to different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR determines the K_f.

For operation in the automatic flow control mode, the same procedure was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at rated power and flow.

The K_f factors shown in Figure 3.11-3, are conservative for Cooper operation because the operating limit MCPR's are greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

References for Bases 3.11.B, 3.11.C, 3.11.D

1. "Cooper Nuclear Station Channel Inspection and Safety Analyses with Bypass Holes Plugged," NEDO-21072, October 1975.

2. Licensing Topical Report, General Electric Boiling Water Reactor, Generic Reload Fuel Application, (NEDE-24011-P), (most current approved submittal).

3. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, February 1973 (NEDO-10802).

4. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDO-20566, dated January 1976.

5. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Unit 1," (most current approved submittal).

6. April 18, 1978 letter from J. M. Pilant (NPPD) to G. E. Lear (NRC).

7. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," NEDO-24154, Volumes 1, 2 and 3, October 1978.

8. Letter, R. H. Buckholz (GE) to P. S. Check (NRC), "ODYN Adjustment Methods for Determination of Operating Limits," January 19, 1981.

9. Letter (with attachment) R. H. Buckholz (GE) to P. S. Check (NRC), "Response to NRC Request for Information on ODYN Computer Model," September 5, 1980.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 70 TO LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

1.0 INTRODUCTION

Nebraska Public Power District (the licensee) requested amendments to the Technical Specifications for the Cooper Nuclear Station (CNS) by letter dated March 5, 1981. The amendments are associated with core Reload Number 6, Cycle 7 operation.

2.0 CORE RELOAD NUMBER 6

2.1 Introduction

By letter(1) dated March 5, 1981, the Nebraska Public Power District (the licensee) requested amendment to the Technical Specifications appended to Operating License DPR-46 for Cooper Nuclear Station (CNS). The proposed changes relate to the refueling of CNS. This reload involves the replacement of 32 exposed 7x7 fuel assemblies and 80 exposed 8x8 assemblies with an equivalent number of fresh, two water rod, P8x8R fuel assemblies designed and fabricated by the General Electric Company (GE). In support of this reload application for CNS, the licensee has submitted a supplemental reload licensing document(2) prepared by GE and proposed plant Technical Specification changes.(3)

The descriptions of the nuclear and mechanical design of the fresh P8x8R fuel assemblies the exposed 8x8R fuel assemblies and the exposed standard 8x8 fuel assemblies, which were used in connection with prior CNS reloads, are contained in GE's generic licensing topical report(4) for BWR reloads. Reference 4 contains a complete set of references to other GE topical reports which describe GE's BWR reload methodologies for the nuclear, mechanical, thermal-hydraulic, transient and accident analysis calculations. Information addressing the applicability of these methods to reload cores containing a mixture of 7x7, 8x8, 8x8R and P8x8R fuel is also contained in Reference 4. Portions of the plant-specific data, such as operating conditions and design parameters used in transient and accident calculations, have also been included in the topical report.

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Our safety evaluations[6,7] of GE's generic reload licensing topical report and supplement concluded that the nuclear and mechanical design of the 8x8R and P8x8R fuel and GE's analytical methods for the nuclear, thermal-hydraulic and transient and accident calculations, as applied to mixed cores containing different fuel types are acceptable.

As part of our evaluations[6] of Reference 4, we found the cycle-independent input data for the reload transient and accident analyses for CNS to be acceptable. The supplementary cycle-dependent information and input data are provided in Reference 2, which follows the format and content of Appendix A of Reference 4. Finally, the licensee has changed the initial core pressure used in the transient analyses from 1045 psia to 1035 psia, to reflect actual plant operating data.

As a result of the staff's generic evaluations[6,7] of a substantial number of safety considerations relating to the use of P8x8R reload fuel in mixed core loadings with 7x7, 8x8 and 8x8R fuel, only a limited number of additional review items are included in this evaluation of Cycle 7 of CNS. These items include the plant and cycle-specific input data and safety analysis results presented in Reference 3, those items identified in our evaluation[6] as requiring special consideration during reload reviews, and the proposed Technical Specification changes.[3]

2.2 Evaluation

2.2.1 Nuclear Characteristics

Reload 6 consists of 112 new P8x8R fuel bundles with bundle average enrichments of 2.83 and 2.65 wt% U-235. The remainder of the 548 fuel assembly reconstituted core will consist of irradiated 7x7, 8x8, 8x8R and P8x8R fuel assemblies exposed during earlier cycles. The assumed cycle exposure has increased from 17,110 MWd/t for Reload 5 to 17,441 MWd/t for Reload 6. The reference core loading for Reload 6 is shown in Figure 1 of Reference 2 will result in quarter core symmetry, which is consistent with previous cycles.

The reload application follows the procedure described in Reference 4. We have reviewed this application and the consequent Technical Specification changes. The transient analysis input parameters provided in Section 6 of Reference 2 are typical for BWRs and are acceptable. Core wide transient analysis results are given for the limiting transients and the required operating limit values for MCPR are given for each fuel type. The revised MCPR limits are required by the reload and they are acceptable.

2.2.2 Thermal Hydraulics

Fuel Cladding Integrity Safety Limit MCPR

As stated in Reference 4, for BWR cores which reload with GE's P8x8R fuel, the allowable minimum critical power ratio (MCPR) resulting from either core-wide or localized abnormal operational transients is equal to 1.07. When meeting this MCPR safety limit during a transient, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition.

The 1.07 safety limit minimum critical power ratio (SLMCPR) to be used for Cycle 7 is unchanged from the SLMCPR previously approved for Cycle 6. The basis for this safety limit is addressed in Reference 4, while our generic approval of the limit is given in References 6 and 7.

Operating Limit MCPR

Various transient events can reduce the MCPR from its normal operating level. To assure that the fuel cladding integrity safety limit MCPR will not be violated during any abnormal operational transient, the most limiting transients have been reanalyzed for this reload by the licensee, in order to determine which event results in the largest reduction in the minimum critical power ratio. These events have been analyzed for the exposed 7x7, 8x8 and 8x8R fuels and for the fresh P8x8R fuel. Addition of the largest reductions in critical power ratio (Δ CPR) to the safety limit MCPR establishes the operating limits for each fuel type.

The transients evaluated were the limiting pressure and power increase transient (in this case, the load rejection transient without turbine bypass to the main condenser), the limiting coolant temperature decrease transient (loss of feedwater heater), the feedwater controller failure transient, the control rod withdrawal error transient and the fuel loading error transient. Initial conditions and transient input parameters as specified in Sections 6 and 7 of Reference 2 were assumed.

The nonpressurization transients were analyzed using the methods described in Reference 4. As per a letter to all BWR Licensees from the NRC[8] all operating BWRs reload submittals with General Electric analyses received after February 1, 1981 are requested to have the limiting transients (overpressurization) recalculated with the ODYN code. The proposed technical specification changes for Cycle 7 will incorporate the requirements of ODYN (Option B) and new MCPR limits associated with these new analyses.

The calculated system responses and ΔCPRs for the above listed operational transients and conditions have been analyzed by the licensee. Results for 100% power/100% flow core conditions were as follows:

TABLE 1

Transient	Exposure	ϕ (% NBR)	Q/A (% NBR)	P_{SL} (psig)	P_v (psig)	ΔCPR 7x7	ΔCPR 8x8	ΔCPR P8x8R & 8x8R
Load Rejection w/o Bypass	BOC-EOC	501.5	122.3	1179	1213	0.14	0.19	0.21
Loss of 100°F Heater Feedwater	--	123	121.7	1022	1069	0.12	0.14	0.14
Feedwater Controller Failure	--	314.4	119.1	1135	1172	0.09	0.13	0.14
Rod Withdrawal Error @10% RBM Set Point							0.16	0.08

The operating MCPR is dependent on the cycle average scram time (τave) and the adjusted analysis scram time (τB). If $\tau B > \tau ave$, the limiting event is a rotated fuel bundle (fuel loading error). The operating MCPR(s) are obtained from the fuel loading error analysis. However, if $\tau ave > \tau B$ the limiting event is load rejection without bypass and the ODYN option B method is used to determine the limiting MCPR(s).

Based on the most limiting transient the licensee has proposed the following operating limit MCPRs (OLMCPR's).

TABLE 2

	7x7	8x8	8x8R	P8x8R
Fuel Loading Error	1.23	1.24	1.24	1.24
ODYN OLMCPR's*	T/S Fig 3.11-2a	T/S Fig 3.11-2b	T/S Fig 3.11-2c	T/S Fig 3.11-2d

*Determined from ODYN Option B

Since the higher OLMCPR obtained from the analyses will preclude violation of the safety limit MCPR of 1.07 in the event of any anticipated operational occurrence, we find these limits to be acceptable.

2.2.3 Overpressure Analyses

For Cycle 7 the licensee has reanalyzed the limiting pressurization event to demonstrate the ASME boiler and pressure vessel code requirements are met. The overpressure analysis for the MSIV closure with high flux scram has been performed in accordance with the requirements of Reference 7.

The sensitivity of peak vessel pressure to failure of one safety valve has also been evaluated. The acceptance criteria for this event is that the calculated peak transient pressure not exceed 110% of design pressure, i.e., 1375 psig. The reanalysis shows that the peak pressure at the bottom of the reactor vessel does not exceed 1290 psig for worst case end-of-cycle conditions, even when assuming the effects of one failed safety valve. Therefore, the limiting overpressure as analyzed by the licensee is acceptable.

2.2.4 Thermal Hydraulic Stability

The results of the thermal hydraulic stability analysis (Reference 2) shows that the channel hydrodynamic and reactor core decay ratios at the natural circulation - 105% rod line intersection (which is the least stable physical attainable point of operation) are within the stability limit. Decay ratio for Reload 6 was 0.80 as compared to 0.78 for Reload 5. Since operation in the natural circulation mode will be prohibited by the Technical Specifications, there will be added margin to the stability limit and this is acceptable to the staff. This predicted decay ratio is below the 1.0 ultimate performance limit decay ratio proposed by General Electric.

The staff has expressed generic concerns regarding reactor core thermal-hydraulic stability at the least stable reactor condition. This condition could be reached during an operational transient from high power if the plant were to sustain a trip of both recirculation pumps without a reactor trip. The concerns are motivated by increasing decay ratios as equilibrium fuel cycles are approached and as reload fuel designs change. The staff concerns relate to both the consequences of operating with a decay ratio of 1.0 and the capability of the analytical methods to accurately predict decay ratios. The General Electric Company is addressing these staff concerns through meetings, topical reports and a stability test program. It is expected that the test results and data analysis, as presented in a final test report, will aid considerably in resolving the staff concerns.

2.3 Physics Startup Testing

Several of the key reload safety analysis inputs and results can be assured via preoperational testing. In order to provide this assurance, the licensee will perform a series of physics startup tests, which was described in Reference 9. This test program was submitted previously in connection with the Cycle 5 reload. Our Cycle 5 review found this program to be acceptable. A written report, describing the results of the physics startup tests, will also be provided by the licensee for staff review following completion of the Cycle 7 tests.

2.4 Technical Specifications

The licensee has submitted proposed changes to the Cooper Technical Specification (3). The effects of these changes are to change the MCPR limits to make them consistent with the values presented in Table 2 of this evaluation. Based on the analysis results, these changes to the MCPR limits in the Technical Specifications are found to be acceptable.

3.0 ENVIRONMENTAL CONSIDERATIONS

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR Section 51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: May 22, 1981

REFERENCES

1. Nebraska Public Power District letter (J. Pilant) to USNRC (T. Ippolito) dated March 5, 1981.

2. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Unit 1, Reload 6," Y1003J01A17, January 1981.

3. "Proposed Changes to the Cooper Nuclear Station Technical Specification" appearing as an Enclosure to the NPPD letter to USNRC dated March 5, 1981.

4. "General Electric Boiling Water Reactor Generic Reload Fuel Application," NEDE-24011-P-A-1 - July 1979.

5. General Electric BWR Thermal Analysis Basis (GETAB): "Data Correlation and Design Application," General Electric Company, BWR Systems Department, November 1973 (NEDO-10958).

6. USNRC letter (D. Eisenhut) to General Electric (R. Gridley) dated May 12, 1978.

7. USNRC letter (T. Ippolito) to General Electric (R. Gridley) dated April 16, 1979.

8. NRC letter (D. Eisenhut) to all BWR licensees, dated November 4, 1980.

9. Nebraska Public Power District letter (J. Pilant) to USNRC (T. Ippolito) dated April 16, 1979.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-298

NEBRASKA PUBLIC POWER DISTRICT

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 70 to Facility Operating License No. DPR-46, issued to Nebraska Public Power District (the licensee), which revised the Technical Specifications for operation of the Cooper Nuclear Station, located in Nemaha County, Nebraska. The amendment is effective as of its date of issuance.

The amendment modifies the Technical Specifications to: (1) permit operation of the facility during Cycle 7 with 112 new P8x8R fuel assemblies designed and fabricated by General Electric Company (GE) replacing an equivalent number of 7x7 and 8x8 irradiated fuel assemblies and (2) revise limits based on transient and accident analysis for Cycle 7 core loading which included use of the ODYN code for the analyses of overpressurization transients.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR Section 51.5(d)(4), an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

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For further details with respect to this action, see (1) the application for amendment dated March 5, 1981, (2) Amendment No. 70 to License No. DPR-46, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D. C. and at the Auburn Public Library, 118 - 15th Street, Auburn, Nebraska 68305. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C., 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland this 22nd day of May 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing