December 9, 1986

Docket No. 50-298

Mr. George A. Trevors
Division Manager - Nuclear Support
Nebraska Public Power District
Post Office Box 499
Columbus, Nebraska 68601

Dear Mr. Trevors

The Commission has issued the enclosed Amendment No. 106 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station. This amendment consists of changes to the Technical Specifications in response to your application dated September 17, 1986 (Change No. 29).

The amendment changes the Technical Specifications to reflect the Cycle 11 reload.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

Original Signed by
William O. Long, Project Manager
BWR Project Directorate #2
Division of BWR Licensing

Enclosures:
1. Amendment No. 106 to License No. DPR-46
2. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION

Docket File	SNorris	BGrimes	OPA
NRC PDR	WLong	TBarnhart (4)	LFMB
Local PDR	OGC - Bethesda	WJones	Plant File
EButcher	LJHarmon	DVassallo	RBernero
NThompson	ELJordan	ACRS (10)	JPartlow

OFFICIAL RECORD COPY
*Previously Concurred

DBL:PD#2	DBL:PD#2	OGC - Bethesda*	DBL:PD#2
SNorris:*	WLong*		DMuller*
12/03/86	12/03/86	12/04/86	12/09/86

Mr. George A. Trevors
Nebraska Public Power District

Cooper Nuclear Station

cc:
Mr. G. D. Watson, General Counsel
Nebraska Public Power District
Post Office Box 4999
Columbus, Nebraska 68601

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

Cooper Nuclear Station
ATTN: Mr. G. Horn, Division
Manager of Nuclear Operations
Post Office Box 98
Brownville, Nebraska 68321

Director
Nebraska Department of Environmental
Control
Post Office Box 94877
State House Station
Lincoln, Nebraska 68509

Mr. William Siebert, Commissioner
Nemaha County Board of Commissioners
Nemaha County Courthouse
Auburn, Nebraska 68305

Resident Inspector
U.S. Nuclear Regulatory Commission
Post Office Box 218
Brownville, Nebraska 68321

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

H. Ellis Simmons, Director
Division of Radiological Health
Department of Health
301 Centennial Mall, South
Post Office Box 95007
Lincoln, Nebraska 68509

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. 106
License No. DPR-46

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

   A. The application for amendment by Nebraska Public Power District (the licensee) dated September 17, 1986 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 Specification

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 106, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 9, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 106

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indiciated by marginal lines.

Pages

210
212f
214b
217

LIMITING CONDITIONS FOR OPERATION

3.11 FUEL RODS

Applicability

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

During steady state power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figure 3.11-1 for two recirculation loop operation. For single-loop operation the values in these curves are reduced by 0.84 for 7x7 fuel; 0.86 for 8x8 fuel; 0.77 for 8x8R, P8x8R, and BP8x8R fuel. If at any time during steady state operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR 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.

B. Linear Heat Generation Rate (LHGR)

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR as calculated by the following equation:

$$LHGR_{max} \leq LHGR_d \, [1 - \{(\Delta P/P)_{max}(L/LT)\}]$$

$LHGR_d$ = Design LHGR = G KW/ft.

$(\Delta P/P)_{max}$ = Maximum power spiking penalty = N

SURVEILLANCE REQUIREMENTS

4.11 FUEL RODS

Applicability

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power.

B. Linear Heat Generation Rate (LHGR)

The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% rated thermal power.

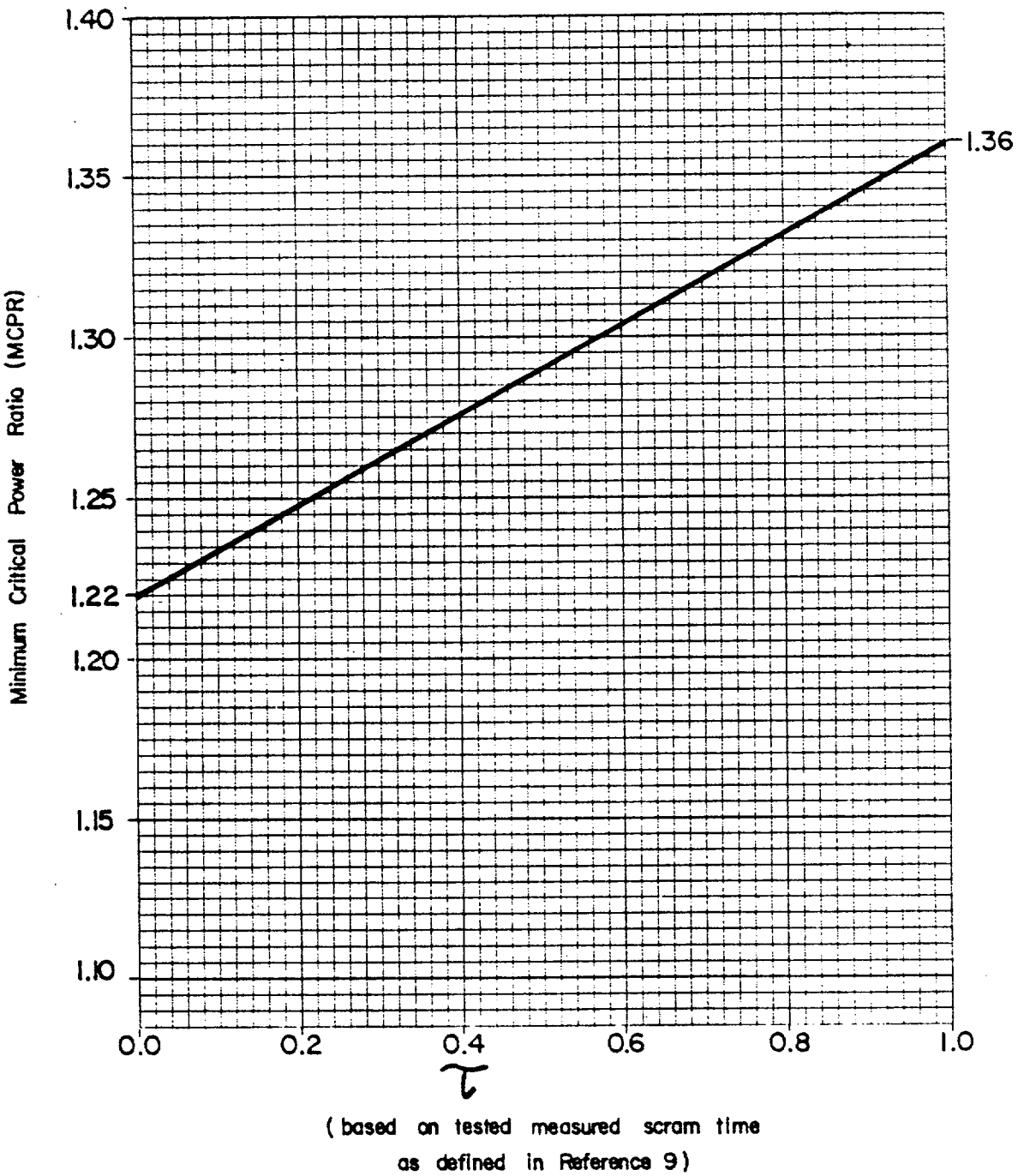

Figure 3.11-2e P8x8R & BP8 x 8R Fuel
(BOC to EOC-1000 MWd/ST)

Amendment No. 93, 106

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 as described in Reference 1.

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

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

2. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Unit 1," (applicable reload document).

3-8. Deleted

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.

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/BP8x8R (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. 106 TO FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By letter dated September 17, 1986 (Ref. 1), Nebraska Public Power District (the licensee) made application to modify the Cooper Nuclear Station (CNS) technical specifications to accommodate the Cycle 11 reload. The letter contained an attachment describing proposed technical specification changes (revised pages 210, 212f, 214b and 217). The licensee also provided a supplemental reload licensing submittal prepared by the General Electric Company dated May 1986 (Ref. 2) which describes the results of the engineering and reload licensing analyses. The description of the Cycle 11 core is given below.

2.0 EVALUATION

2.1 Description of the Cycle 11 Core

The Cooper Nuclear Station Cycle 11 core will consist of 548 fuel assemblies of which 396 are from previous cycles and 152 are new. The new fuel assemblies are of type BP8DRB283 (barrier fuel) which has been approved for use at CNS (Amendment 93) but has not previously been used at CNS. The core for Cycle 11 consists of the 152 new barrier fuel assemblies; 176 type P8DRB283 assemblies (28 from Cycle 10, 60 from Cycle 9, 56 from Cycle 8, and 32 from Cycle 7); and 220 type P8DRB265L assemblies (88 from Cycle 10, 56 from Cycle 9, 56 from Cycle 8, and 20 from Cycle 7).

2.2 Fuel Mechanical Design

The 152 new General Electric (GE) fuel assemblies to be loaded in Cycle 11 are of type BP8DRB283 barrier fuel which exhibits the same design specification parameters 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 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 3 and is therefore acceptable for CNS.

2.3 Nuclear Design

The nuclear design of the Cycle 11 core has been performed with standard General Electric reload methods and techniques which are described in GESTAR II (Reference 4). The results of the analyses are given in Reference 2 in standard GESTAR II format. The shutdown margin of the new core meets the technical specification requirement that the core be at least .38% Δk subcritical in the most reactive condition when the highest worth control rod is fully withdrawn and all other rods are fully inserted. For CNS Cycle 11, GE calculated that the k_{eff} with the strongest rod out is equal to .985 at beginning of the cycle, which for this cycle is the core burnup providing minimum shutdown margin (1.5% Δk).

The standby liquid control system is capable of bringing the reactor from full power to a cold shutdown condition assuming none of the withdrawn control rods is inserted. The 600 ppm boron concentration will bring the reactor subcritical to k_{eff} = .964 at 20° C xenon free condition (Ref. 2). Since these results have been obtained by previously approved methods and fall within the expected range, we conclude that the nuclear design of the Cycle 11 reload core is acceptable.

2.4 Thermal-Hydraulic Design

The thermal-hydraulic design for CNS has been performed with the methodology described in GESTAR II (Reference 4) and the results are given in Reference 2. The parameters used for the analysis are those approved in Reference 4.

The objective of the review is to confirm that the thermal-hydraulic design of the core has been accomplished using acceptable methods, and provides an acceptable margin of safety from conditions which could lead to fuel damage during normal and anticipated operational transients, and is not susceptible to uncontrolled power oscillations due to thermal-hydraulic instability.

The review includes the following areas: (1) safety limit minimum critical power ratio (MCPR), and (2) operating limit MCPR. These are described below.

2.5 Thermal-Hydraulic Stability

The CNS Cycle 11 reload was not analyzed for thermal-hydraulic instabilities. In Amendment 94 the CNS technical specifications were amended to implement improved stability monitoring requirements consistent with GE Service Information Letter 380. In accordance with the staff Safety Evaluation (Ref. 5) of the General Electric Topical Report NEDE-24011 Revision 6, Amendment 8, the facility is exempted from the requirement to perform a cycle specific stability analysis.

2.6 Transient and Accident Analyses

Transient and accident analysis methods are the approved General Electric methods described in Reference 4. These are the same methods that have been used in previous cycles for CNS and they are acceptable for Cycle 11.

A safety limit MCPR has been imposed to assure that 99.9 percent of the fuel rods in the core are not expected to experience boiling transition during normal and anticipated operational transients. As stated in Reference 5, the approved safety limit MCPR is 1.07. The safety limit MCPR of 1.07 is used for Cycle 11 operation.

The most limiting events have been analyzed by the licensee to determine which event could potentially induce the largest reduction (ΔCPR) in the initial critical power ratio. The ΔCPR values given in Section 10 Reference 2 are plant specific values calculated by the approved methods including ODYN methods. The calculated ΔCPRs are adjusted to reflect the calculational uncertainties by employing the conversion methods described in Reference 7. The operating limit MCPR (OLMCPR) values are determined by adding the adjusted ΔCPRs to the safety limit MCPR. Section 12 of Reference 2 presents the Cycle 11 MCPR values of both the pressurization and non-pressurization transients. The maximum MCPR value in Section 12 is the operating limit MCPR for the cycle and must be bounded by the technical specification (TS). The value of operating limit MCPR is determined by the limiting transients, Rod Withdrawal Error (RWE), Feedwater Controller Failure (FWCF) and Load Rejection Without Bypass (LRWBP). The analysis of these events for CNS, via the ODYN Option B approach, provide new Cycle 11 technical specification values of OLMCPR as a function of average scram time, τ. The proposed Cycle 11 TS change for OLMCPR is given in Figure 3.11-2e for BOC to EOC -1000 MWd/ST. That is linear as OLMCPR = $.14\,\tau + 1.22$, however, the Cycle 10 TS is

$$\text{OLMCPR} = \begin{cases} 1.22 \text{ when } \tau \leq .38 \\ .225\,(\tau - .38) + 1.22 \text{ when } \tau \geq .38 \end{cases}$$

therefore, the proposed Cycle 11 TS based on the analysis for the new fuels bounds the Cycle 10 TS values, and is acceptable.

The limiting overpressurization event, the main steam isolation valve (MSIV) closure with flux scram, analyzed with standard GESTAR II methods gave results for peak dome and vessel pressure well under required limits. These are acceptable methodologies and results. LOCA analyses, using approved methodologies and parameters (Reference 4), were performed to provide MAPLHGR values for the new reload fuel assemblies (BP8DRB283). These analyses and results are acceptable.

A cycle specific rod drop accident analysis has been performed for Cycle 11 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.

2.7 Technical Specifications

The changes to be made to the technical specifications are as follows:

A. Page 210 (Limiting Condition for Operation - Average Planar Linear Heat Generation Rate) - The specification would be revised to include BP8x8R fuel among the types of fuel for which the APLHGR reduction factor applies.

B. Page 212f - Figure 3.11-2e - Minimum Critical Power Ratio (MCPR) vs Tau (based on tested measured scram time as defined in Reference 8) for P8x8R and BP8x8R Fuel (BOC to EOC - 1000 MWd/ST). This figure is the result of new analyses.

C. Page 214b - Reference No. 9 (Reference 8) to be added to the bases of Section 3.11 which was inadvertently deleted by Amendment No. 94.

D. Page 217 - (Major Design Features - Reactor) - The description would be revised to indicate that BP8x8R fuel is used.

Each of these is discussed below.

2.7.1 APHLGR Reduction Factor

During single-loop operation, an APLHGR reduction factor is applied for each fuel design installed in accordance with Amendment 94. At the time Amendment 94 was issued BP8x8R fuel was not installed and the reduction factor was thus not included. The appropriate factor for BP8x8R fuel is 0.77, the same as for 8x8R and P8x8R fuel and would be added by this amendment. This is acceptable.

2.7.2 MCPR Specification Format - Figure 3.11-2e

P8x8R and BP8x8R Fuel BOC to EOC - 1000 MWd/ST vs τ: The cycle Minimum Critical Power Ratio (MCPR) as a function of the parameter τ is presented in curve form. This curve is changed to reflect the new transient analyses as previously discussed. The change is acceptable.

2.7.3 Reference No. 9 That Defines τ (Tau) Used in Figure 3.11-2

This reference was previously in technical specifications but was inadvertently deleted by Amendment No. 94. Therefore, the addition to page 214b is acceptable.

2.7.4 Major Design Features - Reactor

Section 5.2.A of the Technical Specifications specifies the number and type of fuel assemblies in the core and would be revised to indicate that barrier fuel BP8x8R is included. This is acceptable.

3.0 ENVIRONMENTAL CONSIDERATIONS

This amendment changes a requirement with respect to 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 the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the 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 the amendment.

4.0 CONCLUSION

As a result of our review, which is described above we conclude that the proposed reload and Technical Specification changes are acceptable. This conclusion is based on the following:

1. Previously approved analysis methods and techniques are employed.

2. The results of the transients and accidents which are affected by the reload are in conformance with the applicable regulation in our standard review criteria and therefore, are acceptable for Cycle 11.

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

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 Contributor: T. Haung

Dated: December 9, 1986

REFERENCES

1. Letter, L. C. Kuncl, Nebraska Public Power District, to D. R. Muller (NRC) dated September 17, 1986.

2. "Supplemental Reload Licensing Submittal for Cooper Nuclear Power Station Unit 1, Reload 10," General Electric Company Report 23A4781 Class 1, May 1986.

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

4. GESTAR II, General Electric Standard Application for Reactor Fuel, NEDE24011f-A-7, dated August 1985.

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

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

7. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," GE Report NEDE-24154-P, October 1978.

8. 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.