December 22, 1992

Docket No. 50-298

Mr. Guy R. Horn
Nuclear Power Group Manager
Nebraska Public Power District
Post Office Box 499
Columbus, Nebraska 68602-0499

Dear Mr. Horn:

SUBJECT: COOPER NUCLEAR STATION - AMENDMENT NO. 156 TO FACILITY OPERATING LICENSE NO. DPR-46 (TAC NO. M82637)

The Commission has issued the enclosed Amendment No. 156 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station (CNS). The amendment consists of changes to the CNS Technical Specifications (TS) in response to your application dated November 15, 1991.

The amendment removes the Rod Sequence Control System from the TS and reduces the Low Power Setpoint for the Rod Worth Minimizer from 20 percent to 10 percent.

A copy of our related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,
ORIGINAL SIGNED BY
Harry Rood, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 156 to License No. DPR-46
2. Safety Evaluation

cc w/enclosures:
See next page

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DATE	11/25/92	11/25/92	11/27/92	12/1/92	12/8/92

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

December 22, 1992

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cc w/enclosures:
See next page

Mr. Guy R. Horn
Nuclear Power Group Manager

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. 156
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 November 15, 1991, 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 license 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.

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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 Appendix A, as revised through Amendment No. 156, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John T. Larkins

John T. Larkins, Director
Project Directorate IV-1
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 22, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 156

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 pages are identified by Amendment number and contain vertical lines indicating the area of change.

REMOVE PAGES	INSERT PAGES
18	18
93	93
94	94
95	95
96	96
97	97
100	100
101	101
101a	101a
216b2	216b2
216b4	216b4

2.1 Bases: (Cont'd)

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit yet allows operating margin that reduces the possibility of unnecessary scrams.

b. APRM Flux Scram Trip Setting (Refuel or Start & Hot Standby Mode)

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the Safety Limit, 25 percent of rated. The margin is adequate to accomodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedure backed up by the Rod Worth Minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the Safety Limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This change can occur when reactor pressure is greater than Specification 2.1.A.6.

3.3 REACTIVITY CONTROL

Applicability:

Applies to the operational status of the control rod system.

Objective:

To assure the ability of the control rod system to control reactivity.

Specification:

A. Reactivity Limitations

1. Reactivity margin - core loading

A sufficient number of control rods shall be operable so that the core could be made subcritical in the most reactive condition during the operating cycle with the strongest control rod fully withdrawn and all other operable control rods fully inserted.

4.3 REACTIVITY CONTROL

Applicability:

Applies to the surveillance requirements to the control rod system.

Objective:

To verify the ability of the control rod system to control reactivity.

Specification:

A. Reactivity Limitations

1. Reactivity margin - core loading

Sufficient control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate, with a margin of 0.38% $\Delta k/k$, that the core can be made subcritical at any time in the subsequent fuel cycle with the analytically determined strongest operable control rod fully withdrawn and all other operable rods fully inserted.

3.3.A (cont'd.)

2. Reactivity margin - inoperable control rods

a. Control rods which cannot be moved with control rod drive pressure shall be considered inoperable. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure the reactor shall be brought to a shutdown condition within 48 hours unless investigation demonstrates that the cause of the failure is not due to a failed control rod drive mechanism collet housing.

b. The control rod directional control valve for inoperable control rods shall be disarmed electrically.

c. Control rods with scram times greater than those permitted by Specification 3.3.C.3 are inoperable, but if they can be inserted with control rod drive pressure they need not be disarmed electrically.

d. Control rods with a failed "Full-in" or "Full-out" position switch may be considered operable if the actual rod position is known. These rods must be moved in sequence to their correct positions (full in on insertion or full out on withdrawal).

e. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable.

f. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met. In addition, during reactor power operation, inoperable control rods shall be separated by at least two control rod cells.

If this Specification cannot be met the reactor shall not be started, or if at power, the reactor shall be brought to a shutdown condition within 24 hours.

4.3 (cont'd)

2. Reactivity margin - inoperable control rods

a. Each partially or fully withdrawn operable control rod shall be exercised one notch at least once each week, when operating above 30% power. This test shall be performed at least once per 24 hours when operating above 30% power in the event power operation is continuing with three or more inoperable control rods or in the event power operation is continuing with one fully or partially withdrawn rod which cannot be moved and for which control rod drive mechanism damage has not been ruled out. The surveillance need not be completed within 24 hours if the number of inoperable rods has been reduced to less than three and if it has been demonstrated that control rod drive mechanism collet housing failure is not the cause of an immovable control rod.

b. Deleted.

c. Once per week, check the status of the pressure and level alarms for each accumulator.

3.3.B (cont'd)

2. The control rod drive housing support system shall be in place during reactor power operation or when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

3.a. Deleted.

b. Deleted.

c. Whenever the reactor is in the startup or run modes below 10% rated power the Rod Worth Minimizer shall be operable or a second licensed operator or other qualified employee shall verify that the operator at the reactor console is following the control rod program. Reactor startup shall not be initiated more frequently than once per calendar year with the RWM inoperable.

d. Deleted.

4.3.B (cont'd)

c. During each refueling outage observe that any drive which has been uncoupled from and subsequently recoupled to its control rod does not go to the overtravel position.

2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.

3a. Deleted.

b. Prior to the start of control rod withdrawal towards criticality and prior to attaining 10% rated power during rod insertion while shutting down, the capability of the Rod Worth Minimizer (RWM) to properly fulfill its function shall be verified by the following checks:

3.3.B.3 (cont'd)

e. If Specification 3.3.B.3.c cannot be met, the reactor shall not be started, or if the reactor is in the run or start-up modes at less than 10% rated power, it shall be brought to a shutdown condition immediately.

4. Control rods shall not be withdrawn for startup unless at least two source range channels have an observed count rate equal to or greater than three counts per second.

5. Operation with a Limiting Control Rod Pattern (for Rod Withdrawal Error, RWE)

a. A Limiting Control Rod Pattern for RWE exists when either:

1) Core thermal power is ≥ 30% and < 90% of rated power and the MCPR is less than 1.70, or

2) Core thermal power is ≥ 90% of rated power and the MCPR is less than 1.40.

b. During operation with a Limiting Control Rod Pattern for RWE either:

1) Both rod block monitor (RBM) channels shall be operable, or

2) With one RBM channel inoperable, control rod withdrawal shall be blocked within 24 hours, or

3) With both RBM channels inoperable, control rod withdrawal shall be blocked until operability of at least one channel is restored.

4.3.B.3.b (cont'd)

1) The correctness of the Banked Position Withdrawal Sequence input to the RWM computer shall be verified.

2) The RWM computer on line diagnostic test shall be successfully performed.

3) Proper annunciation of the selection error of at least one out-of-sequence control rod in each fully inserted group shall be verified.

4) The rod block function of the RWM shall be verified by withdrawing the first rod as an out-of-sequence control rod no more than to the block point.

c. When required, the presence of a second licensed operator or other qualified employee to verify the following of the correct rod program shall be verified.

4. Prior to control rod withdrawal for startup, verify that at least two source range channels have an observed count rate of at least three counts per second.

5. Operation with a Limiting Control Rod Pattern (for Rod Withdrawal Error, RWE)

During operation when a Limiting Control Rod Pattern for RWE exists and only one RBM channel is operable, an instrument functional test of the RBM channel shall be performed prior to withdrawal of the control rod(s). A Limiting Control Rod Pattern for RWE is defined by Specification 3.3.B.5.

3.3 (cont'd)

C. Scram Insertion Times

1. The average scram insertion time, based on the deenergization of the scram pilot valve solenoids as time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.90
50	2.0
90	3.50

2. The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two-by-two array shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	3.71

4.3 (cont'd)

C. Scram Insertion Times

1. After each refueling outage all operable rods shall be scram time tested from the fully withdrawn position with the nuclear system pressure above 800 psig. This testing shall be completed prior to exceeding 40% power. During all scram time testing below 10% power, the Rod Worth Minimizer shall be operable or a second licensed operator or other qualified employee shall verify that the operator at the reactor console is following the control rod program.

2. At 16-week intervals, 10% of the operable control rod drives shall be scram timed above 800 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

cannot be moved with drive pressure. If the rod is fully inserted and then disarmed electrically, it is in a safe position of maximum contribution to shutdown reactivity. If it is disarmed electrically in a non-fully inserted position, that position shall be consistent with the shutdown reactivity limitation stated in Specification 3.3.A.1. This assures that the core can be shutdown at all times with the remaining control rods assuming the strongest operable control rod does not insert. An allowable pattern for control rods valved out of service, which shall meet this Specification, will be determined and made available to the operator.

If damage within the control rod drive mechanism and in particular, cracks in drive internal housings, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress assisted intergranular corrosion have occurred in the collet housing of drives at several BWRs. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed collet housing and requiring increased surveillance after detecting one stuck rod will assure that the reactor will not be operated with a large number of rods with failed collet housings.

B. Control Rod

1. Control rod drop accidents as discussed in the USAR can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. Absence of such response to drive movement could indicate an uncoupled condition. Rod position indication is required for proper function of the Rod Worth Minimizer (RWM).

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in Section III-8.2 of the USAR and the safety evaluation is given in Section III-8.4. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated, since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. The Rod Worth Minimizer (RWM) restrict withdrawals and insertions of control rods to prespecified sequences. These sequences are established such that the drop of any in-sequence control rod or control rod segment (i.e., one or more notches) would not cause the reactor to sustain a power excursion resulting in a peak fuel enthalpy in excess of 280 cal./gm. An enthalpy of 280 cal./gm. is well below the level at which rapid fuel dispersal could occur (i.e., 425 cal./gm.). Primary system damage in this accident is not possible unless a significant amount of fuel is rapidly dispersed. Ref. Subsections III-6.6 and XIV-6.2 of the USAR and Reference 1.

   In performing the function described above, the RWM is not required to impose any restrictions at core power levels in excess of 10% of rated. Material in the cited references shows that it is impossible to reach 280 calories per gram in the event of a control rod drop occurring at power greater than 10%, regardless of the rod pattern. This is true for all normal and abnormal patterns including those which maximize the individual control rod worth.

   At power levels below 10% of rated, abnormal control rod patterns could produce rod worths high enough to be of concern relative to the 280 calories per gram rod drop limit. In this range the RWM constrains the control rod sequences and patterns to those which involve only acceptable rod worths.

   The RWM provides automatic supervision to assure that out of sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences. It serves as a backup to procedural control on control rod sequences, which limits the maximum reactivity worth of control rods. In the event that the RWM is out of service, when required, a second licensed operator or other qualified technical plant employee can manually fulfill the control rod pattern conformance functions of this system.

   The function of the RWM makes it unnecessary to specify a license limit on rod worth to preclude unacceptable consequences in the event of a control rod drop. At low powers, below 10%, this system forces adherence to acceptable rod patterns. Above 10% of rated power, no constraint on rod pattern is required to assure that rod drop accident consequences are acceptable. Control rod pattern constraints above 10% of rated power are imposed by power distribution requirements as defined in Section 3.3.B.5 of these Technical Specifications. Power level for automatic cutout of the RWM function is sensed by feedwater and steam flow.

   Functional testing of the RWM prior to the start of control rod withdrawal at startup, and prior to attaining 10% rated thermal power during rod insertion while shutting down, will ensure reliable operation and minimize the probability of the rod drop accident.

   The Reduced Notch Worth Procedure for control rod withdrawal allows CNS to take advantage of the Banked Position Withdrawal Sequence (BPWS) (Ref. 4). The BPWS has the advantage of having been proven statistically to have such low individual control rod worths that the possibility of a control rod drop accident (CRDA), which exceeds the 280 cal/gm peak fuel enthalpy limit, is precluded (Ref. 1).

The Reduced Notch Worth Procedure is programmed into the RWM. In the pre-checkerboard pattern (100% to 50% control rod density), the RWM will enforce the Reduced Notch Worth Procedure.

4. The Source Range Monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirements of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8}% of rated power used in the analyses of transients cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

3.22 SPECIAL TESTS/EXCEPTIONS (CONT'D)

2. Deleted.

3. RHR System

The RHR system may be aligned in the shutdown cooling mode with at least one shutdown cooling mode loop OPERABLE while performing the Shutdown Margin Demonstration.

4. Containment Systems

Primary containment is not required while performing the Shutdown Margin Demonstration when reactor water temperature is equal to or less than 212°F.

B. Training Startup

1. LPCI System

The LPCI System is required to be operable with the exception that the RHR system may be aligned in the shutdown cooling mode while performing training startups at atmospheric pressure at power levels less than 1% of rated thermal power.

4.22 SPECIAL TESTS/EXCEPTIONS (CONT'D)

B. Training Startup

The reactor vessel shall be verified to be unpressurized and the thermal power verified to be less than 1% of rated thermal power at least once per hour during training startups.

A. Shutdown Margin Demonstration

Performance of shutdown margin demonstrations requires additional restrictions in order to ensure that criticality does not occur. Additional surveillance requirements ensure that shutdown margin requirements and individual rod worths do not exceed values assumed in the safety analysis. Since power levels attained during the demonstration are kept below the level of significant heat addition, the residual heat removal system can remain aligned in the shutdown cooling mode.

B. Training Startup

Specification 3.22.B provides for the performance of training startups without realigning the residual heat removal system from the shutdown cooling mode to the LPCI mode. Power levels during training startups are kept below the level of significant heat addition.

This exception is made in order to minimize contaminated water discharge to the radioactive waste disposal system.

C. Physics Tests

An exception is made to primary containment integrity during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on the system at this time, thus greatly reducing the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the rod worth minimizer would limit control worth such that a rod drop would not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offer a sufficient barrier to keep off-site doses well below 10CFR100 limits.

D. Startup Test Program

Relief from the oxygen concentration specifications is necessary in order to provide access to the primary containment during the initial startup and testing phase of operation. Without this access the startup and test program could be restricted and delayed.

The recirculation flow exception permits reactor criticality under no-flow conditions and is required to perform certain startup and physics tests while at low thermal power levels.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 156 TO FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By letter dated November 15, 1991 (Reference 1), Nebraska Public Power District (the District, NPPD, or the licensee) submitted a request for changes to the Technical Specifications (TS) for Cooper Nuclear Station (CNS). The requested changes would remove the rod sequence control system (RSCS) from the TS and reduce the rod worth minimizer (RWM) low-power set point (LPSP) from its current power level of 20 percent to a power level of 10 percent. The changes will enable the licensee to disable the RSCS for the unit and thereby improve reactor startup and controlled shutdown operations.

2.0 DISCUSSION

The RSCS restricts rod movement to minimize the individual worth of control rods to lessen the consequences of a Control Rod Drop Accident (CRDA). Control rod movement is restricted through the use of rod select, insert, and withdrawal blocks. The RSCS is a hardwired (as opposed to a computer controlled), redundant backup to the RWM. It is somewhat independent of the RWM in terms of direct inputs and outputs but the two systems are similar and compatible and have the same intent. The RSCS and RWM are designed to monitor and block when necessary operator control rod selection, withdrawal and insertion actions, and thus assist in preventing significant control rod pattern errors which could lead to a control rod with a high reactivity worth (if dropped). A significant pattern error is one of several abnormal events, all of which must occur to have a CRDA which might exceed fuel energy density limit criteria for the event. The RSCS was designed only for possible mitigation of the CRDA and is active only during low power operation (currently generally less than 10 or 20 percent power) when a CRDA might be significant. It provides rod blocks on detection of a significant pattern error. The RSCS does not prevent a CRDA. A similar pattern control function is also performed by the RWM, a computer-controlled system. All reactors having an RSCS also have an RWM.

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In August 1986, the Boiling Water Reactor Owner's Group (BWROG), in cooperation with the General Electric Company, proposed Amendment 17 to GESTAR II (References 2 and 3), which would eliminate the requirement for the RSCS and retain the RWM with lowered setpoint for turn-off (during startup) or turn-on (during shutdown) from 20 to 10 percent. The NRC staff reviewed the proposed amendment and concluded that the proposed changes were acceptable. However, the staff's generic approval of Amendment 17 to GESTAR II imposed several additional requirements which must be met by licensees desiring to implement the proposed changes for specific plants. The staff safety evaluation and the additional requirements were provided in an attachment to Reference 4.

The additional requirements that must be met by licensees are:

(1) The TS should require provisions for minimizing operations without use of the RWM.

(2) The occasional necessary use of a second operator replacement should be strengthened by a utility review of relevant procedures, related forms and quality control to assure that the second operator provides an effective and truly independent monitoring process. A discussion of this review should accompany the request for RSCS removal.

(3) Rod patterns used should be at least equivalent to Banked Position Withdrawal Sequence (BPWS) patterns in order to reduce potential maximum rod worth.

3.0 EVALUATION

The CNS licensee has proposed several changes to the TS to implement the removal of the RSCS, to lower the RWM LPSP from 20 to 10 percent, and to meet the three additional requirements imposed by Reference 4.

With regard to requirement (1), above, the licensee states that "In accordance with the NRC's safety evaluation accepting Amendment 17 to GESTAR II, the District has reviewed its control rod movement procedures to ensure verification of control rod movement during RWM inoperability, and has incorporated provisions in the proposed Technical Specification changes to minimize reactor startup with the RWM inoperable." Specifically, TS 3.3.B.3.c has been revised to state that "Reactor startup shall not be initiated more frequently than once per calendar year with the RWM inoperable." This increased administrative control of the RWM follows the pattern of previously approved RWM TS for BWR 3 operation (discussed in Reference 4) and previous reviews of RSCS removal (e.g., Limerick - see Reference 5). These measures have been found to provide the desired improvement in the reliability of the system. The NRC staff has reviewed the licensee's proposed TS changes that implement these operational changes and concludes that they are acceptable, and therefore finds that requirement (1) of Reference 4 has been adequately met.

With regard to requirement (2), above, the licensee states that "In accordance with the NRC's safety evaluation accepting Amendment 17 to GESTAR II, the District has reviewed the CNS operating procedures to verify that adequate controls are in place to ensure an independent verification of correct control rod movement sequence takes place during rod movement with the RWM unavailable. This review has determined that all CNS procedures which govern control rod movements require that while the reactor operating at or below the RWM LPSP, a second licensed operator or other qualified employee must verify conformance to the correct control rod movement sequence when the RWM is inoperable." The NRC staff has reviewed the licensee's discussion of its procedure review and concludes that these procedures provide a suitable, independent check on the rod patterns, and are acceptable. On this basis the staff finds that requirement (2) of Reference 4 has been adequately met.

With regard to requirement (3), above, the licensee states that "CNS employs the Banked Position Withdrawal Sequence (BPWS) control rod movement pattern. The BPWS is a method by which control rods are inserted and withdrawn such that incremental control rod worth are maintained at low values, thereby mitigating the consequences of the CRDA in the startup and low power operating ranges. The BPWS is enforced through the RWM which prevents withdrawing an out-of-sequence control rod more than one notch past the pre-programmed limit." The use of BPWS at CNS was previously approved by the staff in Reference 6. The NRC staff has reviewed the licensee's statements regarding use of the BPWS at CNS and finds them acceptable. On this basis the staff concludes that requirement (3) of Reference 4 has been adequately met.

In summary, the NRC staff has reviewed the proposed TS changes to remove the Rod Sequence Control System from the CNS TS and reduce the Rod Worth Minimizer Low-Power Set Point from its current power level of 20 percent to a power level of 10 percent. The NRC staff has previously reviewed and found acceptable the generic technical justification prepared by the General Electric Company for implementing these changes (Reference 4). The staff approval of these changes for specific plants is conditional on the licensee meeting the three additional requirements defined in Reference 4. The staff review of the proposed changes to the CNS TS concludes that the licensee's application and the proposed changes meet the three additional requirements detailed in Reference 4. Accordingly, the TS changes proposed in Reference 1 are acceptable to the staff and are hereby approved.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Nebraska State official was notified of the proposed issuance of the amendment. The State official had no comment.

5.0 ENVIRONMENTAL CONSIDERATION

The 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 and changes in surveillance requirements. The NRC 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 (57 FR 30251). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 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.

6.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

(1) Letter and enclosures from G. R. Horn, Nebraska Public Power District, to USNRC, dated November 15, 1991, "Proposed Change No. 86 to Technical Specifications, Rod Sequence Control System Removal/RWM Setpoint Reduction, Cooper Nuclear Station, NRC Docket No. 50-298, DPR-46."

(2) Letter and enclosures from T. A. Pickens, BWR Owner's Group, to G. Lainas, NRC, dated August 15, 1986, "Amendment 17 to GE Licensing Topical Report NEDE-24011-P-A."

(3) NEDE-24011-P-A-9, September 1988, "General Electric Standard Application for Reactor Fuel," (GESTAR II).

(4) Letter and enclosures from A. Thadani, NRC, to J. Charnley, General Electric Company, dated December 27, 1987, "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A, Revision 8, Amendment 17."

(5) Amendment 17 to Facility Operating License No. NPF-39, Philadelphia Electric Company, Limerick Generating Station, Unit 1, Docket No. 50-352, dated March 22, 1989.

(6) Amendment 117 to Facility Operating License No. DPR-46, Nebraska Public Power District, Cooper Nuclear Station, Docket No. 50-298, dated February 23, 1988.

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