

December 23, 2004

Mr. Randall K. Edington
Vice President-Nuclear and CNO
Nebraska Public Power District
P. O. Box 98
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION - ISSUANCE OF AMENDMENT
RE: CONTAINMENT ISOLATION LOGIC CHANGE FOR REACTOR VESSEL
WATER LEVEL (TAC NO. MC3320)

Dear Mr. Edington:

The Commission has issued the enclosed Amendment No. 209 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station (CNS). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated May 27, 2004, as supplemented by letter dated September 28, 2004.

The amendment would revise the CNS TS. Specifically it would lower the reactor vessel water level at which the reactor water cleanup (RWCU) system isolates, secondary containment isolates, and the control room emergency filter system starts. General Electric Service Information Letter (SIL) No. 131 discussed problems that result from isolation of the RWCU and start of the standby gas treatment system, in conjunction with isolation of secondary containment. The SIL recommended that the vessel water level at which these actions occur be lowered, thereby eliminating these problems and the resulting unnecessary complications with scram recovery.

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

Sincerely,

/RA/

Michelle C. Honcharik, Project Manager, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosures: 1. Amendment No. 209 to DPR-46
2. Safety Evaluation

cc w/encls: See next page

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 Michelle C. Honcharik, Project Manager, Section 1
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NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 209
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 May 27, 2004, as supplemented by letter dated September 28, 2004, 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.

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. 209, are hereby incorporated in the license. The Nebraska Public Power District shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance and shall be implemented upon startup in Operating Cycle 23.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Michael K. Webb, Acting Chief, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications

Date of Issuance: December 23, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 209

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Replace the following pages of the Appendix A Technical Specifications with the enclosed revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE

3.3-52

3.3-57

3.3-63

INSERT

3.3-52

3.3-57

3.3-63

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 209 TO

FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By application dated May 27, 2004 (Reference 1), as supplemented by letter dated September 28, 2004 (Reference 2), Nebraska Public Power District (NPPD, the licensee), requested changes to the Technical Specifications (TS) for Cooper Nuclear Station (CNS). The supplement dated September 28, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on June 22, 2004 (69 FR 34702).

The proposed changes would revise the CNS TS Tables 3.3.6.1-1, 3.3.6.2-1, and 3.3.7.1-1. Specifically it would lower the reactor vessel water level at which the reactor water cleanup (RWCU) system isolates, secondary containment isolates, and the control room emergency filter system (CREFS) starts. General Electric (GE) Service Information Letter (SIL) No. 131 (Reference 3) discussed problems that result from isolation of the RWCU and start of the standby gas treatment (SGT) system, in conjunction with isolation of secondary containment. The SIL recommended that the vessel water level at which these actions occur be lowered, thereby eliminating these problems and the resulting unnecessary complications with scram recovery.

SIL-131 implementation was previously approved by NRC on Monticello license amendment No. 117, dated March 7, 2001. The logic changes recommended in the SIL-131 are consistent with Standard Technical Specification.

2.0 REGULATORY EVALUATION

This safety evaluation (SE) addresses, in part, the impact of the proposed changes on previously analyzed design basis accident (DBA) radiological consequences and the acceptability of the revised analysis results. The regulatory requirements for which the NRC staff based its acceptance are the accident dose guidelines in (1) Section 100.11 of Title 10 of the *Code of Federal Regulations* (10 CFR), as supplemented by applicable sections of Chapter 15 of the Standard Review Plan (SRP), and (2) 10 CFR Part 50 Appendix A, General Design Criterion (GDC) 19, "Control Room," as supplemented by Section 6.4 of the SRP. The

NRC staff also considered relevant information in the CNS updated safety analysis report (USAR) and TS.

The NRC staff used 10 CFR 50.36 to evaluate the acceptability of the proposed changes to lower the reactor vessel water level at which three automatic actions occur: (1) isolation of the RWCU system, (2) isolation of secondary containment, and (3) initiation of the CREFS in TS Tables 3.3.6.1-1, 3.3.6.2-1, and 3.3.7.1-1, respectively. The regulation at 10 CFR 50.36(c)(1)(ii)(A) requires that a limiting safety setting be specified for a variable on which a safety limit has been placed and that the setting be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded.

Regulatory Guide (RG) 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation," describes a method acceptable to the NRC staff for complying with NRC regulations for ensuring that setpoints for safety-related instrumentation are initiated within and remain within the TS limits. RG 1.105, Revision 3 endorses Part 1 of Instrument Society of America (ISA) Standard ISA-S67.04-1994, "Setpoints for Nuclear Safety-Related Instrumentation." The NRC staff utilized the guidance in RG 1.105 and ISA-S67.04-1994 in performing this review.

3.0 TECHNICAL EVALUATION

3.1 Design Basis Background

NPPD proposes to amend the CNS operating license to lower the reactor vessel water level setpoint at which three automatic protective actions occur. The three automatic actions are: (1) closure of RWCU isolation valves, (2) isolation of secondary containment with attendant start of the SGT system, and (3) start of CREFS. The logic for initiating these protective actions involves a minimum of two diverse plant parameters indicative of the plant condition the response is intended to address (See Table 1 below). One of these actuation signals is Reactor Vessel Water Level - Low (also known as "Level 3"). NPPD proposes to use the Reactor Vessel Water Level - Low Low signal (also known as "Level 2") instead. This effectively lowers the reactor vessel water level for isolation of RWCU, isolation of secondary containment, and start of CREFS from ≥ 3 inches (161.19 inches above top of active fuel) to ≥ -42 inches (116.19 inches above top of active fuel), a difference of 45 inches.

The changes to the logic in this proposed amendment were recommended by GE in SIL-131, which states that reactor scrams from power levels greater than 50 percent may result in reactor vessel water level transients below the reactor low water level scram setpoint (Level 3) due to void collapse (shrink). The resulting isolation of RWCU causes problems such as loss of the filter media in the filter-demineralizers and loss of the ability to remove water from the reactor vessel immediately after a scram. These problems complicate scram recovery. SIL-131 also discusses how isolation of secondary containment can cause increased temperature in the reactor building and result in a thermal transient on the building and equipment. Isolation of reactor building normal ventilation results in isolation of the cooling air supply to the reactor recirculation motor generator (RRMG) sets. The RRMG sets may trip due to high air temperature, thereby tripping the reactor recirculation pumps. Tripping these pumps results in loss of forced recirculation flow through the core.

The following plant systems are involved in each of the three proposed changes to TS:

- (1) TS Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation," Function 5.d, reactor water cleanup (RWCU) system isolation on reactor vessel water level.

The RWCU system isolation function is to limit fission product release during and following postulated DBAs. Reactor vessel level is one of the input parameters to the RWCU isolation logic. The RWCU system performs no safety function. The RWCU system removes impurities from the reactor coolant water by continuously removing a portion of the reactor coolant from the bottom head drain. The removed flow is sent through filter-demineralizer and ion exchange processes. The processed fluid is returned to the reactor through the reactor feedwater line.

- (2) TS Table 3.3.6.2-1, "Secondary Containment Isolation Instrumentation," Function 1, reactor vessel water level.

Secondary containment isolation and establishment of vacuum with the SGT system ensures that fission products that leak from primary containment following a DBA are maintained within applicable limits. The reactor vessel water level is one of the input parameters to the secondary containment isolation logic and to the SGT system initiation. The SGT system filters particulates and iodines to reduce the level of airborne contamination released to the environment. Following initiation, both SGT filter train fans start. Upon verification that both subsystems are operating, the redundant subsystem is normally shut down, with the other SGT train in operation.

- (3) TS Table 3.3.7.1-1, "Control Room Emergency Filter System Instrumentation," Function 1, reactor vessel water level.

CREFS is designed to provide a radiological controlled environment to ensure the habitability of the control room for the safety of control room operators under all plant conditions. Reactor water level is one of the input parameters for closing the normal control room air inlet damper and starting the CREFS. CREFS operation ensures that radiation exposure of control room personnel through the duration of any one of the postulated accidents does not exceed the limits set by GDC 19.

Table 1
Current Licensing Basis Actuation Matrix

Protective Action	Actuation Signals
Reactor Water Cleanup System Isolation <i>TS Table 3.3.6.1-1</i>	a. RWCU Flow - High b. RWCU System Space Temperature - High c. SLC System Initiation d. Reactor Vessel Water Level - Low
Secondary Containment Isolation <i>TS Table 3.3.6.2-1</i>	1. Reactor Vessel Water Level - Low 2. Drywell Pressure - High 3. Reactor Building Ventilation Exhaust Plenum Radiation - High

Control Room Emergency Filter Instrumentation	1.	Reactor Vessel Water Level - Low
	2.	Drywell Pressure - High
	3.	Reactor Building Ventilation Exhaust Plenum Radiation - High

TS Table 3.3.7.1-1

In the event of lowered vessel level following a reactor scram, this logic change will reduce the potential for post-scram operator challenges by reducing the potential for unnecessary isolations of RWCU and secondary containment, with attendant initiation of SGT and CREFS. Eliminating post-scram operator challenges will enhance plant safety.

Conforming changes to the Bases are needed to reflect that these three actions will occur at reactor vessel water Level 2 instead of Level 3. These changes were included in Reference 1 for information.

This SE considers those impacts that affect assumptions or inputs in a DBA radiological consequence analysis. NPPD evaluated the impact of these changes and concluded that the proposed changes would not impact the previous radiological consequence analysis results. As such, no re-analyses were deemed necessary.

3.2 Control Rod Drop Accident (CRDA) Analysis

This accident analysis postulates a sequence of mechanical failures that results in the rapid removal (i.e., drop) of a control rod resulting in localized fuel damage. This accident results in the release of radioactive material from the fuel with the reactor coolant pressure boundary, primary containment, and secondary containment initially intact.

NPPD states in Reference 1 that the current analysis for a CRDA at CNS assumes that the CREFS is manually initiated after 24 hours and that changes to the automatic actuation logic for CREFS will not impact the results of the control room habitability analyses.

The CRDA is analyzed assuming that the initial reactor power is at 10 percent and that the reactor pressure vessel (RPV) water level is initially at the normal level as these two assumptions establish a limiting case for the analysis. NPPD states in Reference 1 that although reactor scrams from greater than 50 percent power usually result in a Level 3 trip, it is unlikely that a reactor scram at 10 percent power would result in a Level 3 trip, let alone the proposed Level 2 trip. There is no reactor coolant boundary breach associated with a CRDA. As such, the Level 3 trip is not a mitigating action for a CRDA and the logic change from Level 3 to Level 2 for the RWCU isolation and secondary containment isolation has no effect on the plant response to a CRDA.

The NRC staff considered the licensee's evaluation in conjunction with the radiological consequences analysis discussion in the CNS USAR. Based on its review, the NRC staff concurs with the licensee's assessment that the proposed logic change has no impact on the analyzed radiological consequences of a design basis CRDA.

3.3 Loss-of-Coolant Accident (LOCA) Analysis

A LOCA is a failure of the reactor coolant system that results in the loss of reactor coolant which, if not mitigated, could result in fuel damage including core melt. In the event of an

LOCA, RPV water level will decrease rapidly due to the loss of reactor coolant via spillage through the break location and as the water flashes to steam due to the depressurization of the RPV. The release of the RPV water causes the primary containment pressure to rapidly increase resulting in a Drywell Pressure - High trip. This trip occurs essentially concurrently with the LOCA and would initiate secondary containment isolation and CREFS actuation prior to the Level 3 trip occurring. Since the Drywell Pressure - High trip precedes the Level 3 trip (and the Reactor Building Ventilation Exhaust Plenum Radiation - High trip), the proposed logic change from Level 3 to Level 2 does not alter the CNS plant response to the LOCA.

Isolation of the RWCU system is initiated for a break in the RWCU system to prevent exceeding offsite dose criteria. Drywell Pressure - High is not an actuation signal for RWCU isolation, since much of the RWCU system is located outside of the primary containment. Instead, the RWCU Flow - High and RWCU System Space Temperature trips, as well as the proposed Level 2 trip, would actuate RWCU isolation.

The NRC staff considered the licensee's evaluation in conjunction with the radiological consequences analysis discussion in the CNS USAR. Based on its review, the NRC staff concurs with the licensee's assessment that the proposed logic change has no impact on the analyzed radiological consequences of a design basis LOCA.

3.4 Fuel Handling Accident (FHA) Analysis

The FHA analysis postulates that a spent fuel assembly is dropped during refueling. The kinetic energy developed in this drop is conservatively assumed to be dissipated in the damage to the cladding on 151 fuel rods. The fission product inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released to the reactor building, resulting in a Reactor Building Ventilation Exhaust Radiation - High trip. This trip results in secondary containment isolation and CREFS actuation.

The Reactor Building Ventilation Exhaust Radiation - High trip is unaffected by the proposed lowering of RPV water level. Reactor coolant inventory is unaffected by the FHA. The proposed change to the RPV water level isolation logic signal will have no effect on CNS response to the FHA.

The NRC staff considered the licensee's evaluation in conjunction with the radiological consequences analysis discussion in the CNS USAR. Based on its review, the NRC staff concurs with the licensee's assessment that the proposed logic change has no impact on the analyzed radiological consequences of a design basis FHA.

3.5 Main Steam Line Break (MSLB) Accident Analysis

The MSLB accident considered is the complete severance of a main steam line outside the primary containment. The main steam isolation valves (MSIV) are assumed to isolate the leak. The radiological consequences of a break outside containment will bound the results from a break inside containment. A break outside the secondary containment is assumed since this location is more limiting from a radiological perspective (e.g., no release mitigation by SGT).

CREFS actuation may occur later in the accident sequence, since Level 2 is reached after Level 3. However, this slight delay in CREFS actuation has no impact on the radiological

consequences of a DBA MSLB, because the MSLB radiological consequence analysis does not credit CREFS operation.

A break in the RWCU is not postulated for the DBA MSLB. As such, RWCU isolation is not a factor in the radiological consequence analyses and the proposed change from Level 3 to Level 2 has no impact on the MSLB analysis. Secondary containment isolation is not a factor since the break is assumed to occur outside of the secondary containment. As such, the proposed change from Level 3 to Level 2 has no impact on the radiological consequences of an MSLB.

The NRC staff considered the licensee's evaluation in conjunction with the radiological consequences analysis discussion in the CNS USAR. Based on its review, the NRC staff concurs with the licensee's assessment that the proposed logic change has no impact on the analyzed radiological consequences of a design basis MSLB.

3.6 Allowable Value (AV) Change Evaluation

The proposed amendment will lower from Level 3 to Level 2 the reactor vessel water level at which RWCU isolates, secondary containment isolates (with the attendant start of the SGT system), and CREFS initiates. The AV for Level 3 is 3 inches and Level 2 is -42 inches. These values of reactor vessel water level are relative to a level designated "Instrument Zero," which is 516.75 inches above the bottom of the reactor pressure vessel and 158.19 inches above the top of active fuel. Level 2 is the level at which high pressure coolant injection and reactor core isolation cooling initiate.

During recent reviews of proposed license amendments associated with changes to the TS limiting safety system setting (LSSS), the NRC staff identified a concern regarding the method used by some licensees to determine the TS AVs. AVs are used in the TS as LSSSs, to provide acceptance criteria for determination of instrument channel operability during periodic surveillance testing. The NRC staff's concern relates to one of the three methods for determining the AV as described in ISA recommended practice ISA-RP67.04-1994, Part II, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation."

The regulation at 10 CFR 50.36(c)(1)(ii)(A), states, in part, that where a LSSS is specified for a variable on which a safety limit (SL) has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a SL is exceeded. The analytical limit (AL) is the limit on the process variable at which the instrument loop protective action occurs as assumed in the plant's safety analysis. Protective action at the AL ensures that the SL is not exceeded. The AL, however, does not account for uncertainties associated with the instrument loop. The instrument loop uncertainty is accounted for during the calculation of an instrument loop's trip setpoint.

To verify which method was used at CNS for calculating the TS AVs, the NRC staff requested the licensee to provide additional information related to the instrument setpoint methodology used to calculate the reactor vessel water level AVs. In Reference 2, the licensee stated that CNS uses the GE setpoint methodology, specified by NEDC-31336P-A, "General Electric Instrument Setpoint Methodology," dated September 1996, to determine instrument AVs. This methodology is consistent with Method 2 of the ISA recommended practice ISA-RP67.04.02-2000, "Methodologies for the Determination of Setpoints for Nuclear

Safety-Related Instrumentation," dated January 1, 2000. In addition, the GE setpoint methodology incorporates a "licensing event report avoidance" evaluation that can provide additional margin between the AV and the nominal trip setpoint because GE setpoint methodology recommended in licensee's surveillance procedures that trip setting should be moved further from the analytical limit.

The licensee further stated that calculation of the AVs for both Level 3 and Level 2 were calculated by the GE setpoint methodology. The NRC approved use of the GE setpoint methodology at CNS by License Amendment No. 178, dated July 31, 1998. The licensee determined that the calculation of the AV for reactor vessel water Level 3 and Level 2 was not affected by the proposed change to implement GE SIL-131.

Based on the above discussion, the NRC staff finds that the proposed amendment lowering the reactor vessel water level at which RWCU and secondary containment isolate and CREFS initiates from Level 3 to Level 2 is not subject to the AV "generic concern," because CNS uses Method 2 to determine the TS AVs.

3.7 Evaluation of Proposed TS Changes

3.7.1 TS Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation"

Table 3.3.6.1-1 identifies plant parameters (functions) that cause isolation of primary containment. The table also identifies the plant modes in which the functions are applicable, the number of channels required for each trip system, applicable surveillance requirements (SRs), and the AV for each function that causes primary containment isolation. Function 5 addresses RWCU isolation. NPPD proposes to change Function 5.d, "Reactor Vessel Water Level - Low (Level 3)," to read "Reactor Vessel Water Level - Low Low (Level 2)." The AV of 3 inches would be changed to -42 inches.

As discussed above, NRC staff concurs with the licensee's assessment that the proposed logic change has no impact on the analyzed radiological consequences of DBAs at CNS and that the proposed change is acceptable with regard to accident radiological consequences. Additionally, the NRC staff concurs with the licensee's assessment with respect to implementing the GE SIL-131 recommendation, and the NRC staff finds that the use of ISA-RP67.04-1994 Method 2 to determine the TS AV is in conformance with RG 1.105 and 10 CFR 50.36. Therefore the proposed TS change is acceptable.

3.7.2 Technical Specification Table 3.3.6.2-1, "Secondary Containment Isolation Instrumentation"

Table 3.3.6.2-1 identifies plant parameters that cause isolation of secondary containment. The table also identifies the plant modes in which the functions are applicable, the number of channels required for each trip system, applicable SRs, and the AV for each function that causes secondary containment isolation. NPPD proposes to change Function 1, "Reactor Vessel Water Level - Low (Level 3)," to read "Reactor Vessel Water Level - Low Low (Level 2)." The AV of 3 inches would be changed to -42 inches.

As discussed above, NRC staff concurs with the licensee's assessment that the proposed logic change has no impact on the analyzed radiological consequences of DBAs at CNS and that the

proposed change is acceptable with regard to accident radiological consequences. Additionally, the NRC staff concurs with the licensee's assessment with respect to implementing the GE SIL-131 recommendation, and the NRC staff finds that the use of ISA-RP67.04-1994 Method 2 to determine the TS AV is in conformance with RG 1.105 and 10 CFR 50.36. Therefore the proposed TS change is acceptable.

3.7.3 TS Table 3.3.7.1-1, "Control Room Emergency Filter System Instrumentation"

Table 3.3.7.1-1 identifies plant parameters that cause the actuation of the CREFS. The table also identifies the plant modes in which the functions are applicable, the number of channels required for each trip system, applicable SRs, and the AV for each function that actuates CREFS. NPPD proposes to change Function 1, "Reactor Vessel Water Level - Low (Level 3)," to read "Reactor Vessel Water Level - Low Low (Level 2)." The AV of 3 inches would be changed to -42 inches."

As discussed above, NRC staff concurs with the licensee's assessment that the proposed logic change has no impact on the analyzed radiological consequences of DBAs at CNS and that the proposed change is acceptable with regard to accident radiological consequences. Additionally, the NRC staff concurs with the licensee's assessment with respect to implementing the GE SIL-131 recommendation, and the NRC staff finds that the use of ISA-RP67.04-1994 Method 2 to determine the TS AV is in conformance with RG 1.105 and 10 CFR 50.36. Therefore the proposed TS change is acceptable.

3.8 Technical Conclusion

Based on the review of References 1 and 2, the NRC staff finds that the proposed TS changes on lowering the AVs for RWCU system isolation, secondary containment isolation, and CREFS initiation are in conformance with 10 CFR 50.36 and RG 1.105. The proposed TS changes are therefore acceptable. Additionally, the NRC staff considered the licensee's evaluation in conjunction with the radiological consequences analysis discussion in the CNS USAR. Based on its review, the NRC staff concurs with the licensee's assessment that the proposed logic change has no impact on the analyzed radiological consequences of design basis accidents at CNS. Therefore, the NRC staff finds the proposed license amendment acceptable with regard to the radiological consequences of postulated DBAs.

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 comments.

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. 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

published June 22, 2004 (69 FR 34702). 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.

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 to USNRC from NPPD dated May 27, 2004 (ADAMS Accession No. ML041540109)
2. Letter to USNRC from NPPD dated September 28, 2004 (ADAMS Accession No. ML042780484)
3. General Electric BWR Services Information Letter, SIL-131, dated March 31, 1975 (ADAMS Accession No. ML041770459)

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Date: December 23, 2004

Cooper Nuclear Station

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