UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 2, 2010

Mr. David J. Bannister
Vice President and CNO
Omaha Public Power District
Fort Calhoun Station
444 South 16th St. Mall
Omaha, NE 68102-2247

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: REVISION TO TECHNICAL SPECIFICATION 2.15, TABLE 2-5 FOR SAFETY VALVE ACOUSTIC POSITION INDICATION **(EMERGENCY CIRCUMSTANCES)** (TAC NO. ME3992)

Dear Mr. Bannister:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 265 to Renewed Facility Operating License No. DPR-40 for the Fort Calhoun Station, Unit No. 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated May 31, 2010, as supplemented by letter dated June 1, 2010.

The amendment modifies TS 2.15, "Instrumentation and Control Systems," Table 2-5, Note c to allow a one-time extension of the 7-day allowed outage time for inoperability of Item 4, "Safety Valve Acoustic Position Indication," to allow repair prior to the next entry into Operating Mode 3 (Hot Shutdown) from Operating Mode 4 (Cold Shutdown).

The license amendment is issued under emergency circumstances as provided in the provisions of paragraph 50.91(a)(5) of Title 10 of the *Code of Federal Regulations* due to the time critical nature of the amendment. In this instance, an emergency situation exists in that the proposed amendment is needed to allow the licensee to preclude a plant shutdown.

A copy of the related Safety Evaluation is also enclosed. The safety evaluation describes the emergency circumstances under which the amendment was issued and the final no significant hazards determination. A Notice of Issuance addressing the final no significant hazards determination and opportunity for a hearing associated with the emergency circumstances, will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

Lynnea E. Wilkins, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosures:
1. Amendment No. 265 to DPR-40
2. Safety Evaluation

cc w/encls: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

OMAHA PUBLIC POWER DISTRICT

DOCKET NO. 50-285

FORT CALHOUN STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 265
Renewed License No. DPR-40

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

    A. The application for amendment by the Omaha Public Power District (the licensee), dated May 31, 2010, as supplemented by letter dated June 1, 2010, 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, Renewed Facility Operating License No. DPR-40 is amended by changes as indicated in the attachment to this license amendment, and paragraph 3.B. of Renewed Facility Operating License No. DPR-40 is hereby amended to read as follows:

   B. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 265, 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 and shall be implemented upon issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License No. DPR-40 and Technical Specifications

Date of Issuance: June 2, 2010

ATTACHMENT TO LICENSE AMENDMENT NO. 265

RENEWED FACILITY OPERATING LICENSE NO. DPR-40

DOCKET NO. 50-285

Replace the following pages of the Renewed Facility Operating License No. DPR-40 and the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

License Page

REMOVE	INSERT
-4-	-4-

Technical Specifications

REMOVE	INSERT
2.15 – Page 14	2.15 – Page 14

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or instrument calibration or when associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by operation of the facility.

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is, subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

A. Maximum Power Level

Omaha Public Power District is authorized to operate the Fort Calhoun Station, Unit 1, at steady state reactor core power levels not in excess of 1500 megawatts thermal (rate power).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 265 are hereby incorporated in the license. Omaha Public Power District shall operate the facility in accordance with the Technical Specifications.

C. Security and Safeguards Contingency Plans

The Omaha Public Power District shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Fort Calhoun Station Security Plan, Training and Qualification Plan, Safeguards Contingency Plan," submitted by letter dated May 19, 2006.

TABLE 2-5

Instrumentation Operating Requirements for Other Safety Feature Functions

No.	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Condition
1	CEA Position Indication Systems	1	None	None
2	Pressurizer Level	1	None	Not Applicable
3	PORV Acoustic Position Indication-Direct	1[(a)(c)]	None	Not Applicable
4	Safety Valve Acoustic Position Indication	1[(a)(c)]	None	Not Applicable
5	PORV/Safety Valve Tail Pipe Temperature	1[(d)(b)]	None	Not Applicable

NOTES:

a One channel per valve.
b One RTD for both PORV's; two RTD's, one for each code safety.
c If item 5 is operable, requirements of specification 2.15 are modified for items 3 and 4[i] to "Restore inoperable channels to operability within 7 days or be in hot shutdown within 12 hours."
d If items 3 and 4 are operable, requirements of specification 2.15 are modified for item 5 to "Restore inoperable channels to operability within 7 days or be in hot shutdown within 12 hours."

[i] The requirement of Table 2-5, Note c to restore Safety Valve Acoustic Position Indication in 7 days is extended on a one-time basis. This allows the instrumentation for Functional Unit 4 for pressurizer safety valve RC-142 to be inoperable from June 1, 2010 until the next entry into Mode 3 from Mode 4.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 265 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-40

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT NO. 1

DOCKET NO. 50-285

1.0 INTRODUCTION

By application dated May 31, 2010, as supplemented by letter dated June 1, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML101520198 and MLxxxxxxxxx[1], respectively), Omaha Public Power District (OPPD, the licensee) requested changes to the Technical Specifications (TSs) (Appendix A to Renewed Facility Operating License No. DPR-40) for the Fort Calhoun Station, Unit No. 1.

The proposed amendment would modify TS 2.15,"Instrumentation and Control Systems," Table 2-5, Note c to allow a one-time extension of the 7-day allowed outage time for inoperability of Item 4,"Safety Valve Acoustic Position Indication," to allow repair prior to the next entry into Operating Mode 3 (Hot Shutdown) from Operating Mode 4 (Cold Shutdown).

Currently, the TSs only allow this monitor to be inoperable for 7 days, with a subsequent action to be in Hot Shutdown within 12 hours. As a result of the monthly surveillance test IC-ST-RC-0001, "Functional Test of Acoustic Flow Monitors," conducted May 26, 2010, the licensee declared flow indicator FI-142, associated with safety valve RC-142, inoperable and entered TS 2.15, Table 2-5, Item 4 at 9:35 a.m. Central Daylight Time (CDT). The licensee's troubleshooting activities have determined that components outside of the containment building are not the cause. The licensee's troubleshooting activities inside of the containment building have not yet definitively identified the cause, but a cable failure is suspected. Although repair efforts are continuing, the licensee is not confident that FI-142 can be returned to operability prior to expiration of the current allowed outage time at 9:35 a.m. CDT on June 2, 2010.

2.0 REGULATORY EVALUATION

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TS as part of the license. The Commission's regulatory requirements related to the content of TS are contained in Title 10 of the *Code of Federal Regulations* (10 CFR),

[1] As of the date of issuance of this amendment, the supplemental letter dated June 1, 2010, had not yet been added to ADAMS.

Enclosure 2

Part 50, Section 50.36, "Technical Specifications. The TS requirements in 10 CFR 50.36 include the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) SRs; (4) design features; and (5) administrative controls. The requirements for instrumentation and controls are included in the TS in accordance with 10CFR50.36(c)(2), "Limiting Conditions for Operation."

In a memorandum dated September 18, 1992 (ADAMS Legacy Library Accession No. 9210060362), the Commission approved the NRC staff's proposal in SECY-92-223, "Resolution of Deviations Identified During the Systematic Evaluation Program," not to apply 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," to plants with construction permits prior to May 21, 1971. FCS was licensed for construction prior to May 21, 1971, and at that time committed to the draft General Design Criteria (GDC). The draft GDC, which are similar to Appendix A, "General Design Criteria for Nuclear Power Plants," in 10 CFR Part 50, are contained in Appendix G, "Response to 70 Criteria," of the FCS Updated Safety Analysis Report (USAR).

NRC Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," December 1980.

NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," July 1979.

NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

In its letter dated May 31, 2010, the licensee appropriately identified the following draft GDC as specified in Appendix G to the FCS USAR:

> <u>FCS Design Criterion 12 – Instrumentation and Control Systems</u> states:
>
> Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges.
>
> <u>FCS Design Criterion 16 – Monitoring Reactor Coolant Pressure Boundary</u> states:
>
> Means shall be provided for monitoring the reactor coolant pressure boundary to detect leakage.

3.0 <u>TECHNICAL EVALUATION</u>

3.1 <u>System Description</u>

In its supplemental letter dated June 1, 2010, the licensee stated:

> The FCS reactor coolant system (RCS) is protected against overpressurization by control and protective circuits such as the pressurizer pressure high reactor

trip and by the 2 power-operated relief valves [(PORVs)] (PCV-102-1 and PCV-102-2) and the 2 PSVs [pressurizer safety valves] (RC-141 and RC-142) connected to the top of the pressurizer. Upon opening, these valves discharge steam into the pressurizer quench tank, which condenses and collects the valve effluent. Two independent monitoring systems (acoustic and temperature) exist to alert the operator to the passage of steam or liquid through the PSVs due to valve lift or seat leakage. [Each PSV is equipped with a temperature sensor in the tail pipe of the valve. In addition, the PSV's are equipped with individual acoustic sensors.] The purpose of the PSV acoustic monitor is to provide the operator with information regarding PSV position by detecting [...] acoustic vibrations generated from the steam flowing through the valve and actuat[ing] an alarm when a temperature increase is experienced in the line, as would be the case if the valve released steam.

The acoustic monitors were added to the FCS TSs by Amendment No. 54 [issued January 19, 1981, ADAMS Legacy Library Accession Nos. 8101300609 and 8101300614] to meet the requirements of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations," and NUREG-0737, "Clarification of TMI Action Plan Requirements." With inoperable acoustic position indication, OPPD will utilize the temperature sensor [associated with] PSV RC-142 to identify flow through RC-142. This sensor provides indication and alarm in the control room and indication on the plant computer.

The PSVs discharge into the pressurizer quench tank. The temperature, pressure, and liquid level of this tank are indicated and alarmed in the control room. A change in these parameters would alarm and alert the operator of a PSV discharge condition. Abnormal Operating Procedures (AOP-22) and Emergency Operating Procedures (EOP-03) contain instructions noting that RCS leakage to the pressurizer quench tank is indicated by a rise in tank pressure, temperature, or level and rising or elevated pressure relief line temperatures or flow indication from the relief line acoustic monitors. Operators undergo continuous training on utilization of the AOPs and EOPs.

During monthly surveillance testing, the single fixed impactor is detected by all four acoustic monitor channels as observed on test instrumentation. This is caused by the sensitivity of the accelerometers and the fact that the safety valves and their associated acoustic monitor sensors are in close proximity to each other. Functional checks of the PORVs are detected by all four acoustic monitors. Therefore, if the safety valve associated with the inoperable valve position acoustic monitor channel was to discharge, the remaining three valve acoustic monitor channels would alert operations.

Additionally, both code safety valves are monitored by independent tail pipe temperature sensors that are recorded every two hours by operations. The highest of Pressurizer Safety Valve (RC-14.1 and RC-142) discharge temperatures TIA-135 & TIA-136 and the PORV discharge temperature TIA-134 are logged every two hours on FC-75 Control Room log. These readings are trended by the operators taking them and reviewed once per shift by the Control

Room Supervisor or Shift Manager, [thus, providing the ability of the operators to detect a potentially inoperable temperature sensor.]

These independent temperature loops are function tested on a monthly surveillance. Furthermore, the safety valves discharge into the pressurizer quench tank. Temperature, pressure, and liquid level of the quench tank are calibrated on an 18 month frequency. The quench tank level, pressure, and temperatures have indication and alarms in the control room. These parameters are recorded once every 12 hours.

3.2 Proposed TS Changes

In its letter dated May 31, 2010, the licensee proposes to add a footnote to Note c in TS 2.15, Table 2-5 which currently states:

> If item 5 is operable, requirements of specification 2.15 are modified for items 3 and 4 to "Restore inoperable channels to operability within 7 days or be in hot shutdown within 12 hours."

Revised Note c in TS 2.15, Table 2-5 would state:

> If item 5 is operable, requirements of specification 2.15 are modified for items 3 and 4[i] to "Restore inoperable channels to operability within 7 days or be in hot shutdown within 12 hours."

The TS 2.15, Table 2-5 footnote would state:

> [i] The requirement of Table 2-5, Note c to restore Safety Valve Acoustic Position Indication in 7 days is extended on a one-time basis. This allows the instrumentation for Functional Unit 4 for pressurizer safety valve RC-142 to be inoperable from June 1, 2010 until the next entry into Mode 3 from Mode 4.

3.3 Precedents

In its letter dated May 31, 2010, the licensee states:

> As noted in Sections 2.0 and 3.0 above, precedent for allowing an extended allowed outage time (AOT) for inoperable PSV acoustic position indication was found in Amendment No. 161 for the Donald C. Cook Nuclear Plant (ADAMS Accession No. ML021060546) and in Amendment No. 162 for the Palisades Nuclear Plant (ADAMS Accession No. ML020840096). In addition, precedence for allowing continued operation with an inoperable acoustic monitor on the safety/relief valve tailpipe under an emergency license amendment request is found in Amendment No. 100 (ADAMS ML No. 10110103) for Susquehanna Steam Electric Station. Susquehanna Steam Electric Station is a boiling water reactor.

3.4 Analysis (As provided by the licensee)

As stated in its letter dated June 1, 2010, in accordance with 10 CFR 50.91(a)(5), the licensee provided the following additional information in support of its request:

> No analyses were conducted in support of the proposed amendment. This instrument is not credited in Probabilistic Risk Assessment (PRA) for operator actions to mitigate the consequences of an event.
>
> The PRA was reviewed to determine the risk impact of the failed acoustic monitor. As shown above, it is one of several indications used by the operators to identify a stuck open PORV or PSV. Given the effectiveness of the other indications, inoperability of the acoustic monitor is judged to have a negligible impact upon core damage frequency. Therefore, there is a high level of confidence that the inoperable acoustic monitor has a negligible impact upon safe power operation.
>
> The risk associated with plant transition and shutdown has a higher level of uncertainty. The subject of transition risk was evaluated by CE NPSD- 1021, Rev. 03, "Development of a Methodology for the Evaluation of Transition Risk." It was prepared for the Combustion Engineering Owners Group in January 1997. This report concluded that the total core damage frequency for shutdown to repair a failed safety injection tank (SIT) is 7.49E-08. This risk is dominated by the overall risk of shutting down and starting up the plant.
>
> Considering the risk associated with plant shutdown and restart, and considering the negligible impact of the failed acoustic monitor upon core damage frequency, it is judged that nuclear safety is preserved by continuing power operation.

3.5 NRC Staff Evaluation

The licensee plans to utilize other PORV/PSV acoustic indications as well as individual PSV tail pipe temperatures as backup indication of RCS leakage. The licensee states that the proposed amendment would allow acoustic position indication for PSV RC-142 to be inoperable until the next Mode 3 entry from Mode 4 after June 1, 2010, but does not eliminate permanently this requirement nor does it eliminate the requirement for acoustic position indication for PSV RC-141 or PORV/PSV tail pipe temperature indication. The NRC concludes that the proposed amendment concerning inoperability of PSV acoustic position indication for RC-142 does not adversely affect detection of RCS leakage for compliance with GDC 16. These methods of detecting RCS leakage are still available and, therefore, is acceptable.

In its letter dated May 31, 2010, the licensee states that "the electronics of the acoustic monitors are sufficiently independent so as to allow the affected acoustic monitor (i.e., FI-142) to be isolated without affecting the acoustic monitors that remain OPERABLE."

As mentioned above, the quench tank level, pressure, and temperatures have indication and alarms in the control room. These parameters are recorded once every 12 hours. The licensee states that a change in any of these parameters would alarm and alert the operator of a safety valve discharge condition. If the affected safety valve were to lift, the sensitivity of the remaining operable acoustic monitors is sufficient to detect the lifting of the PSV or substantial flow through the affected safety valve discharge line.

The pressurizer discharge piping acoustic monitors were put in place to provide positive indication of relief valve actuation and reseating in the control room, whereas tailpipe temperature and quench tank parametric indications do not provide this exact function. While the discharge and quench tank parameter indications would provide a delayed indication whether the relief valves had opened and reseated or simply remained open, the primary and most readily indicated parameter to demonstrate whether the reactor coolant pressure boundary has been restored is the primary coolant system pressure. If any of the pressurizer relief valves open and fail to re-seat, the RCS pressure will continue to decrease, thereby alerting operators that an inadvertent valve opening has occurred and has not been mitigated. The NRC staff concludes that the licensee has established reasonable controls and that RCS pressure indications would alert the plant operator as to whether continued pressurizer relief valve leakage or discharge were occurring.

The licensee's proposed TS change request has the impact of extending the allowable outage time for the acoustic monitor position indication for one of the two pressurizer safety valves from 7 days to a maximum of approximately 1 year. Based on the above discussions regarding the existence of backup instrumentation, the NRC staff concludes the proposed TS change are in accordance with 10 CFR 50.36 and are therefore, acceptable.

4.0 EMERGENCY CIRCUMSTANCES

In its letter dated May 31, 2010, as supplemented by letter dated June 1, 2010, the licensee requested that the amendment be treated as an emergency amendment.

The licensee stated in its May 31, 2010, letter that troubleshooting activities inside of the containment building have not yet definitively identified the cause, but a cable failure is suspected. If, as the licensee suspects, a cable failure is the cause, the repair cannot be undertaken while the reactor is at power. Although repair efforts are continuing, the licensee is not confident that FI-142 can be returned to operability prior to expiration of the current allowed outage time at 9:35 a.m. CDT on June 2, 2010.

The license amendment is issued under emergency circumstances as provided in the provisions of paragraph 50.91(a)(5) of Title 10 of the *Code of Federal Regulations* due to the time critical nature of the amendment. The NRC staff evaluated whether the licensee took reasonable action to avoid the situation. Troubleshooting efforts by the licensee confirm that the repairs cannot be made while the plant is at power. Therefore, the NRC staff concludes that the licensee's actions were reasonable and that the emergency situation could not have been avoided.

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION

The Commission may issue license amendments before the expiration of the 60-day period provided that its final determination is that the amendments involve no significant hazards consideration. This amendment is being issued prior to the expiration of the 60-day period. Therefore, a final finding of no significant hazards consideration follows.

The Commission has made a final determination that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment does not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

As required by 10 CFR 50.91(a), in its letter dated May 31, 2010, as supplemented by letter dated June 1, 2010, the licensee has provided its analysis of the issue of no significant hazards consideration which is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The safety valve acoustic position indication does not affect the operation of its associated spring-loaded safety valve. As such, the proposed change does not increase the probability of an accident. The acoustic monitor is only one of the indications used to identify that a safety valve is open. Other indications are available to the operators and alarm in the control room. The acoustic monitor is only one of the indications that the abnormal and emergency procedures direct operators to use to diagnose the opening of a safety valve. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any previously evaluated?

   Response: No.

   The safety valve acoustic position indication does not perform a control or active protection function. It only provides indication. Additional indications are available and alarm in the control room to provide the operator with equivalent information. Because of the diverse indication system, failure or mis-operation of this indicator will not cause an operator to mis-diagnose an event.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

The proposed change results in operators having one less indicator of the position of safety valve RC-142. The operators are provided with other diverse indications which include safety valve discharge temperature, and pressurizer quench tank level, pressure, and temperature. Abnormal and emergency procedures direct the operators to use these indications to determine the status of the safety valves.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Accordingly, the Commission has determined that this amendment involves no significant hazards determination.

6.0 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.

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the 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. 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.

8.0 CONCLUSION

The NRC staff 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.

Principal Contributor: Lynnea Wilkins

Date: June 2, 2010

A copy of the related Safety Evaluation is also enclosed. The safety evaluation describes the emergency circumstances under which the amendment was issued and the final no significant hazards determination. A Notice of Issuance addressing the final no significant hazards determination and opportunity for a hearing associated with the emergency circumstances, will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Lynnea E. Wilkins, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosures:
1. Amendment No. 265 to DPR-40
2. Safety Evaluation

cc w/encls: Distribution via Listserv

DISTRIBUTION:
PUBLIC
LPLIV Reading
RidsAcrsAcnw_MailCTR Resource
RidsNrrDeEicb Resource
RidsNrrDirsItsb Resource
RidsNrrDorlDpr Resource
RidsNrrDorlLpl4 Resource
RidsNrrDssSrxb Resource
RidsNrrPMFortCalhoun Resource
RidsNrrLAJBurkhardt Resource
RidsOgcRp Resource
RidsRgn4MailCenter Resource

ADAMS Accession No. ML101520296 *via telecom **via email

OFFICE	NRR/LPL4/PM	NRR/LPL4/PM	NRR/LPL4/LA	DIRS/ITSB/BC	DSS/SRXB/BC
NAME	LWilkins	AWang	JBurkhardt	RElliott CSchulten for	AUlses**
DATE	6/2/10	6/2/10	6/2/10	6/2/10	6/2/10
OFFICE	DE/EICB/BC	OGC NLO	NRR/LPL4/BC	NRR/LPL4/PM	
NAME	WKemper*	BMizuno	MMarkley	LWilkins	
DATE	6/2/10	6/2/10	6/2/10	6/2/10	

OFFICIAL RECORD COPY