UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 E LAMAR BLVD
ARLINGTON, TX 76011-4511

July 1, 2014

Mr. Lou Cortopassi, Vice President
and Chief Nuclear Officer
Omaha Public Power District
Fort Calhoun Station FC-2-4
P.O. Box 550
Fort Calhoun, NE 68023-0550

SUBJECT: FORT CALHOUN STATION – NRC EXAMINATION REPORT 05000285/2014301

Dear Mr. Cortopassi:

On May 22, 2014, the U.S. Nuclear Regulatory Commission (NRC) completed an initial operator license examination at Fort Calhoun Station. The enclosed report documents the examination results and licensing decisions. The preliminary examination results were discussed on May 22, 2014, with Mr. L. Cortopassi, Vice President and Chief Nuclear Officer, and other members of your staff. A telephonic exit meeting was conducted on June 17, 2014, with Mr. M. Joe, Initial License Training Supervisor, who was provided the NRC licensing decisions.

The examination included the evaluation of four applicants for reactor operator licenses and two applicants for instant senior reactor operator licenses. The license examiners determined that all of the applicants satisfied the requirements of 10 CFR Part 55 and the appropriate licenses have been issued. There were five post examination comments submitted by your staff. Enclosure 1 contains details of this report and Enclosure 2 summarizes post examination comment resolution.

No findings were identified during this examination.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Vincent G. Gaddy, Chief
Operations Branch
Division of Reactor Safety

Docket: 50-285
License: DPR-40

Enclosures:
1. NRC Examination Report 05000285/2014301
2. NRC Post Examination Comment Resolution

cc w/enclosure: Electronic Distribution

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Letter to Lou Cortopassi from Vincent G. Gaddy, dated July 1, 2014

SUBJECT: FORT CALHOUN STATION – NRC EXAMINATION REPORT 05000285/2014301

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ACES (R4Enforcement.Resource@nrc.gov)
MC 0350 Panel Chairman (Anton.Vegel@nrc.gov)
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MC 0350 Panel Member (Michael.Balazik@nrc.gov)
MC 0350 Panel Member (Michael.Markley@nrc.gov)

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000285
License:	DPR-40
Report:	05000285/2014301
Licensee:	Omaha Public Power District
Facility:	Fort Calhoun Station
Location:	Fort Calhoun Station FC-2-4 P.O. Box 550 Fort Calhoun, NE 68023-0550
Dates:	May 19 through May 22, 2014
Inspectors:	S. Garchow, Chief Examiner, Senior Operations Engineer B. Larson, Senior Operations Engineer G. Apger, Operations Engineer
Approved By:	Vincent Gaddy, Chief Operations Branch Division of Reactor Safety

SUMMARY OF FINDINGS

ER 05000285/2014301; 05/19/2014 – 05/22/2014; Fort Calhoun Station; Initial Operator Licensing Examination Report.

NRC examiners evaluated the competency of four applicants for reactor operator licenses and two applicants for instant senior reactor operator licenses at Fort Calhoun Station.

The licensee developed the examinations using NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, Supplement 1. The written examination was administered by the licensee on May 27, 2014. NRC examiners administered the operating tests on May 19-21, 2014.

The examiners determined that all the applicants satisfied the requirements of 10 CFR Part 55, and the appropriate licenses have been issued.

A. NRC-Identified and Self-Revealing Findings

None

B. Licensee-Identified Violations

None

REPORT DETAILS

4. OTHER ACTIVITIES (OA)

4OA5 Other Activities (Initial Operator License Examination)

.1 License Applications

a. Scope

NRC examiners reviewed all license applications submitted to ensure each applicant satisfied relevant license eligibility requirements. Examiners also audited two of the license applications in detail to confirm that they accurately reflected the subject applicant's qualifications. This audit focused on the applicant's experience and on-the-job training, including control manipulations that provided significant reactivity changes.

b. Findings

No findings were identified.

.2 Examination Development

a. Scope

NRC examiners reviewed integrated examination outlines and draft examinations submitted by the licensee against the requirements of NUREG-1021. The NRC examination team conducted an on-site validation of the operating tests.

b. Findings

NRC examiners provided outline, draft examination, and post-validation comments to the licensee. The licensee satisfactorily completed comment resolution prior to examination administration.

NRC examiners determined the written examinations and operating tests initially submitted by the licensee were within the range of acceptability expected for a proposed examination.

.3 Operator Knowledge and Performance

a. Scope

On May 27, 2014, the licensee proctored the administration of the written examinations to all six applicants. The licensee staff graded the written examinations, analyzed the results, and presented their analysis and post examination comments to the NRC on June 6, 2014.

The NRC examination team administered the various portions of the operating tests to all applicants on May 19-22, 2014.

b. Findings

No findings were identified.

All applicants passed the written examination and all parts of the operating test. The final written examinations, and post examination analysis and comments, may be accessed in the ADAMS system under the accession numbers noted in the attachment.

The examination team noted three generic weaknesses associated with applicant performance on the simulator JPM and inplant JPM sections of the operating tests. The applicants displayed a weakness in locating some electrical distribution controls and indications, paralleling AC sources, and calculating the required temperature setpoint for the boric acid batch tank. Copies of all individual examination reports were sent to the facility training manager for evaluation and determination of appropriate remedial training.

.4 Simulation Facility Performance

a. Scope

The NRC examiners observed simulator performance with regard to plant fidelity during examination validation and administration.

b. Findings

No findings were identified.

.5 Examination Security

a. Scope

The NRC examiners reviewed examination security for examination development during both the onsite preparation week and examination administration week for compliance with 10 CFR 55.49 and NUREG-1021. Plans for simulator security and applicant control were reviewed and discussed with licensee personnel.

b. Findings

No findings were identified.

4OA6 Meetings, Including Exit

The chief examiner presented the preliminary examination results to Messrs. L. Cortopassi, Vice President and Chief Nuclear Office, J. Lindsey, Training Director, and other members of the staff on May 22, 2014. A telephonic exit was conducted on June 17, 2014, between Mr. S. Garchow, Chief Examiner, and Mr. M. Joe, Initial License Training Supervisor.

The licensee did not identify any information or materials used during the examination as proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

S. Shea, Operations Training Manager
D. Dryden, Nuclear Training Instructor
J. Koske, Nuclear Training Instructor
R. Fisher, Nuclear Training Instructor

NRC Personnel

J. Kirkland, Senior Resident Inspector

ADAMS DOCUMENTS REFERENCED

Accession No. ML14170A107 - FINAL WRITTEN EXAMS (withheld two years)
Accession No. ML14181A030 - FINAL OPERATING TEST
Accession No. ML14170A040 - POST EXAM ANALYSIS-COMMENTS

NRC Resolution to the Fort Calhoun Station Post Examination Comments

A complete text of the licensee's post examination analysis and comments can be found in ADAMS under Accession Number ML14170A040.

RO QUESTION # 46

COMMENT: The licensee recommended accepting distractor 'D' as the only correct answer. This question asks how long the station batteries would last following a station blackout assuming the non-essential DC loads are removed. The correct answer as originally graded, four hours, was based on the value from the FSAR and was determined to be the minimum time as required by NRC regulations. The actual site-specific design calculation indicates the batteries would last at least eight hours under the most bounding conditions.

NRC RESOLUTION: The NRC agrees with the licensee's recommendation to accept distractor 'D' as the only correct answer for Question #46. This is based on distractor 'D' being derived from the plant-specific design calculations, whereas distractor 'B' represents the generic NRC required value.

RO QUESTION # 47

COMMENT: The licensee recommended accepting distractor 'D' as the only correct answer. This question was focused on how battery voltage will respond following a loss of its associated battery charger. The original answer, battery voltage would remain stable and then drop off at a rapid rate, ignored the initial response when the loss of the battery charger occurred. When the battery charger is lost, there would be an initial voltage drop. Thus, distractor 'D,' an initial short drop off, then lower at a linear rate, is correct.

NRC RESOLUTION: The NRC agrees with the licensee's recommendation to change the correct answer from 'B' to 'D.' A review of the plant-specific battery capacity discharge test data indicates there would be an initial short voltage drop followed by a linear voltage decrease. This makes distractor 'D' the only correct answer.

RO QUESTION # 58

COMMENT: The licensee recommended accepting distractor 'C' as the only correct answer. This question asked how a "suspect" Core Exit Thermocouple (CET) alarm would be displayed on the Quality Safety Parameter Display System (QSPDS) computer and if it would be used in the QSPDS calculations. The original correct distractor stated the "suspect" value would be used. However, it was determined that if the value was more than one standard deviation from the norm, the input would not be used in any QSPDS calculations. The "suspect" CET value provided in the stem was more than one standard deviation from the norm and, therefore, would not be used. Thus distractor 'C', the "suspect" CET value would not be used, is correct.

NRC RESOLUTION: The NRC agrees with the licensee's recommendation to change the correct answer from 'D' to 'B.' A review of the Chauvenet Criterion using standard deviations from the normal samples supports the value provided in the stem would be more than one standard deviation from the norm and would not be used. This makes distractor 'B' the only correct answer.

SRO QUESTION # 83

COMMENT: The licensee recommended accepting distractors ‘A’ and ‘B’ as both being correct. This question provides the applicants with a set of plant conditions involving a dropped fuel assembly during fuel handling operations and asks what procedure to enter as well as what action to take. The correct answer had the correct procedure to enter and listed closing at least one Personnel Access Lock (PAL) door as the required action. However, tripping the Ventilation Isolation Actuation Signal (VIAS) is also listed as an action later in the same procedure. Distractor ‘A’ also had the correct procedure to enter and listed tripping VIAS as an operator action making it also correct.

NRC RESOLUTION: The NRC agrees with the licensee's recommendation to accept both distractors ‘A’ and ‘B’ as correct. Both distractors list the same correct procedure and both distractors list an operator action as defined by this procedure. This makes both distractors, ‘A’ and ‘B’ correct.

SRO QUESTION # 98

COMMENT: The licensee recommended deleting this question because it has no correct answer. This question provides the applicants with a set of plant conditions involving a Steam Generator Tube Rupture (STGR) and asks what the maximum dose would be at the site boundary for the first two hours of the SGTR. It was determined during the exam review the “correct” distractor (taken from the FSAR), 25 REM TEDE, was based on the NRC limits contained in 10 CFR 50.67 and not the site-specific analysis. The site-specific analysis as calculated in OPPD Calculation FC06820, Revision 1, “Site Boundary and Control Room Dose following a Steam Generator Tube Rupture Accident Using Alternate Source Terms”, concludes the maximum site boundary dose would be 1.0 REM. Because none of the distractors list 1.0 REM as a possible answer, there is no correct answer and the question should be deleted.

NRC RESOLUTION: The NRC agrees with the licensee's recommendation to delete Question 98 because it has no correct answer. The two-hour dose, as provided from the FSAR, is actually the regulatory limit as defined by 10 CFR 50.67 and not the site-specific value. Since the site-specific calculated value is 1.0 REM and none of the distractors list this value, there is no correct answer. Therefore, this question is not valid.