UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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January 18, 2008

Charles D. Naslund, Senior Vice
President and Chief Nuclear Officer
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SUBJECT: CALLAWAY PLANT - NRC EXAMINATION REPORT 05000483/2007301

Dear Mr. Naslund:

On November 30, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an examination at Callaway Plant. The enclosed report documents the examination findings, which were discussed on December 7, 2007, with Mr. David G. Lantz.

The examination included the evaluation of 8 applicants for reactor operator licenses, 3 applicants for instant senior operator licenses and 1 applicant for an upgrade senior operator license. The written and operating examinations were developed using NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9. The license examiners determined that all 12 applicants satisfied the requirements of 10 CFR Part 55, and the appropriate licenses have been issued.

No findings of significance 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/

Ryan E. Lantz, Chief
Operations Branch
Division of Reactor Safety

Docket: 50-483
License: NPF-30

Enclosure:
NRC Examination Report 05000483/2007301

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EXAMINATION REPORT
U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Dockets:	50-483
Licenses:	NPF-30
Report :	05000483/2007301
Licensee:	AmerenUE
Facility:	Callaway Plant
Location:	Junction Highway CC and Highway O Fulton, Missouri
Dates:	November 26 through December 7, 2007
Inspectors:	M. S. Haire, Chief Examiner, Operations Branch T. F. Stetka, Senior Operations Engineer S. M. Garchow, Senior Operations Engineer G. W. Apger, Operations Engineer
Approved By:	Ryan E. Lantz, Chief Operations Branch Division of Reactor Safety

SUMMARY OF FINDINGS

ER 05000483/2007301; November 26 through December 7, 2007; Callaway Plant; Initial Operator Licensing Examination Report.

NRC examiners evaluated the competency of 8 applicants for reactor operator licenses, 3 applicants for instant senior operator licenses and 1 applicant for an upgrade senior operator license at Callaway Plant. The facility licensee developed the examinations using NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9. The written examination was administered by the facility on November 21, 2007. NRC examiners administered the operating tests on November 26-30, 2007. The license examiners determined that all 12 applicants satisfied the requirements of 10 CFR Part 55, and the appropriate licenses have been issued.

A. <u>NRC-Identified and Self-Revealing Findings</u>

No findings of significance were identified.

B. <u>Licensee-Identified Violations</u>

None.

REPORT DETAILS

4. OTHER ACTIVITIES (OA)

4OA5 Other Activities (Initial Operator License Examination)

.1 License Applications

a. Scope

The examiners reviewed all 12 license applications submitted by the licensee to ensure the applications reflected that each applicant satisfied relevant license eligibility requirements. The applications were submitted on NRC Form 398, “Personal Qualification Statement,” and NRC Form 396, “Certification of Medical Examination by Facility Licensee.” The 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 of significance were identified.

.2 Operator Knowledge and Performance

a. Examination Scope

On November 21, 2007, the licensee proctored the administration of the written examinations to all 12 applicants. The licensee staff graded the written examinations, analyzed the results, and presented their analysis to the NRC on November 30, 2007.

The NRC examination team administered the various portions of the operating examination to all 12 applicants on November 26-30, 2007. Of the 8 applicants for reactor operator licenses, 5 participated in three dynamic simulator scenarios and 3 participated in two dynamic simulator scenarios; additionally, all 8 participated in a control room and facilities walkthrough test consisting of 11 system tasks, and an administrative test consisting of 4 administrative tasks. The 3 applicants seeking an instant senior operator license participated in three dynamic simulator scenarios, a control room and facilities walkthrough test consisting of 10 system tasks, and an administrative test consisting of 5 administrative tasks. The applicant for an upgrade senior operator license participated in three dynamic simulator scenarios, a control room and facilities walkthrough test consisting of 5 system tasks, and an administrative test consisting of 5 administrative tasks.

b. Findings

All 12 of the applicants passed all parts of the operating test and the written examination. For the written examinations, the reactor operator applicants' average score was 84.7 percent and ranged from 81.3 to 88.0 percent, and the senior operator applicants' average score was 83.5 percent and ranged from 80.0 to 87.0 percent. The overall written examination average was 84.3 percent. The text of the examination questions, the licensee's examination analysis, and the licensee's post-examination comments may be accessed in the ADAMS system under the accession numbers noted in the attachment.

Chapter ES-403 and Form ES-403-1 of NUREG-1021 require the licensee to analyze the validity of any written examination questions that were missed by half or more of the applicants. The licensee conducted this performance analysis for 15 questions that met this criterion and an additional 3 questions with flaws (based on applicant feedback) and submitted an analysis of those 18 questions to the chief examiner. This analysis concluded that 4 of the questions (Examination Questions: 3, 16, 21, 45, and 87) required modification, and the other 14 questions analyzed were psychometrically and technically sound and represented knowledge weaknesses that would be addressed in training.

The licensee's recommendations and the NRC responses follow:

Reactor/Senior Operator Question 3

The question originally read as follows:

Given the following:

- The plant is in Mode 4.
- RHR [residual heat removal] Train "A" is in service.
- RHR Heat Exchanger Bypass Valve EJ FCV-618 is set to maintain 3400 GPM.
- RHR Heat Exchanger outlet valve EJ HCV-606 demand position set at 30%.
- The Instrument Air supply line to RHR Heat Exchanger Bypass Valve EJ FCV-618 becomes severed and is completely detached.
- No other air operated valves are impacted by the failure.

Which ONE (1) of the following describes the RHR system parameter changes from the initial steady state conditions?

	RHR HX Outlet Temp.	Total RHR flow
A.	Higher	Higher
B.	Higher	Lower
C.	Lower	Lower
D.	Lower	Higher

Original proposed answer: C

The licensee recommended accepting two answers, B and C, on the basis of the following information:

(Note: a system drawing can be viewed in the post examination comments file in ADAMS; Figure-1 below is a simplified flow diagram of the RHR heat exchanger and bypass for discussion purposes.)

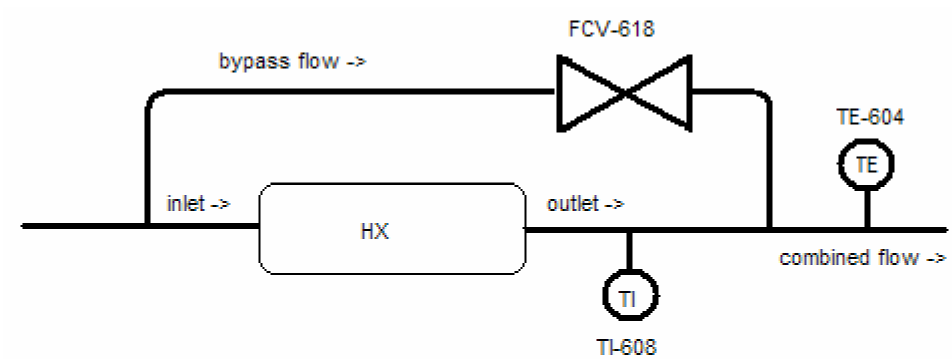

Figure-1

The question supposes Valve FCV-618 completely fails closed (air supply is lost) and asks the applicant to predict both the effect in total RHR flow and the effect on the "RHR HX Outlet Temp." However, the question fails to differentiate between the local temperature indication for "RHR HX Outlet Temp." on TI-608 and the control room indication for "RHR HX Outlet Temp." as detected by TE-604. An analysis of the effect on the system of FCV-618 failing closed easily concludes that "Total RHR Flow" will be lower, while the temperature indicated on TI-608 will be higher and the temperature indicated on TE-604 will be lower. Therefore, since the question failed to define whether "RHR HX Outlet Temp." was intended to refer to TI-608 or TE-604, the applicant could have reasonable assumed either temperature indication and therefore could have reasonably and correctly chosen either Answer B or C.

NRC Response for 3: The NRC agrees with the licensee's recommendation to accept both answers B and C since both answers are simultaneously correct depending on which temperature indication the applicant assumes is meant by "RHR HX Outlet Temp."

Reactor/Senior Operator Question 16

The licensee made an editorial change to answer B during the administration of the exam on the basis of the following information:

During administration of the written exam, an applicant asked whether there was a typographical error in answer B. Answer B, as given on the exam, read: "Main Feed Header Pressure Channel, ABPT0508, fails Low..." This was a typographical error in that ABPT0508 refers to the main steam header pressure channel rather than the main feed header pressure channel. Answer B should have read: "Main Feed Header Pressure Channel, ABPT0507, fails Low..." This error was corrected during the administration of the exam and all candidates were made aware of the error and its correction.

NRC Response for #16: The NRC agrees with the licensee's correction of answer B during written exam administration so that it referred to ABPT0507 instead of incorrectly referencing ABPT0508.

Reactor/Senior Operator Question 21

The question originally read as follows:

Given the following:
The plant is in Mode 1.
A leak exists in the "A" EDG Fuel Oil Storage Tank.
Level indicates as follows:

0845 86%
0900 85%
0915 83%
0930 81%
0945 77%

From the choices below, what was the EARLIEST time that "A" EDG was inoperable in accordance with technical specifications?

A. 0900
B. 0915
C. 0930
D. 0945

Original proposed answer: C

The licensee recommended accepting two answers, C and D, on the basis of the following information:

The question asks for the earliest time that the "A" emergency diesel generator (EDG) was inoperable due to the EDG fuel oil storage tank level being too low.
Procedure OSP-NE-0001A, "Standby Diesel Generator 'A' Periodic Tests," Revision 28,

Attachment 3, allows the operator to perform the tank level surveillance using either of two available indicators; however, the two indicators do not register the same level in percent when the tank is at the operability limit of 80,900 gallons. If the Computer Point JEL0003, "DG Fuel TK A LEV," is used, then the acceptance criteria is “>79.6%” (a 79.6 percent level on this indicator corresponds to 80,900 gallons in the tank). Using this indication would make Answer D correct. Additionally, if Indicator JELIT0005, "Emerg F.O. STOR TK A LEV IND XMTR," is used, then the acceptance criteria is "> 82%" (an 82 percent level on this indicator also corresponds to 80,900 gallons in the tank). Using this indication would make Answer C correct. Since the question does not specify which indicator is used for the percent level, the applicant would get Answer C by assuming the level was indicated on Computer Point JEL0003, and the applicant would get Answer D by assuming the level was indicated on Indicator JELIT0005. Therefore, both Answers C and D are correct.

NRC Response for 21: The NRC agrees with the licensee's recommendation to accept both answers C and D as correct.

Reactor/Senior Operator Question 45

The question originally read as follows:

Given the following:

- The plant is operating at 100% power.
- A failure of the controlling input to the Pressurizer Pressure Master Controller caused actual pressurizer pressure to increase to 2285 psig.
- The Pressurizer Pressure Master Controller has been placed in MANUAL.

Which ONE (1) of the following describes the action required to return pressure to 2235 psig?

A. Decrease the controller output
B. Increase the controller output
C. Lower the pressure setpoint adjustment
D. Raise the pressure setpoint adjustment

Original proposed answer: B

The licensee recommended accepting only Answer C on the basis of the following information:

The answer key had proposed Answer B, “Increase the controller output,” as the correct answer. However, in accordance with Procedure OOA-BB-00002, "Actuations vs. Controller Output," this was an error since controller output should be **lowered** to cause the pressurizer spray valves to open, thus, reducing pressurizer pressure. Therefore, Answer C should be the only answer accepted as correct.

NRC Response for 45: The NRC agrees with the licensee's recommendation to accept only answer C as correct.

Reactor/Senior Operator Question 87

The question originally read as follows:

Chemistry sample has determined the following:

- "A" SI Accumulator boron concentration is 2306 ppm.
- "B" SI Accumulator boron concentration is 2292 ppm.
- "C" SI Accumulator boron concentration is 2307 ppm.
- "D" SI Accumulator boron concentration is 2299 ppm.
- RWST boron concentration is 2358 ppm.

Which ONE (1) of the following describes the impact of this condition in accordance with technical specifications?

A. RWST operability status may not be adequate to counteract the reactivity effects of an uncontrolled RCS cool down. Comply with the actions of TS 3.5.4 to restore RWST boron concentration or initiate a plant shutdown to

B. RWST operability status may adversely affect the assumptions made for transfer to hot leg recirculation following a LOCA. Comply with the actions of TS 3.0.3 and initiate action to restore RWST operability.

C. SI Accumulator operability may not meet the requirements of ECCS acceptance criteria for maintenance of a coolable core geometry. Comply with the actions of TS 3.5.1 to restore SI Accumulator operability or initiate a plant shutdown to Hot Standby.

D. SI Accumulator operability may not meet the requirements to assure subcriticality in a post-LOCA environment. Comply with the actions of TS 3.0.3 and initiate action to restore SI accumulator operability.

Original proposed answer: D

The licensee recommended accepting two answers, C and D, on the basis of the following information:

The Technical Specification Bases for Limiting Condition of Operation (LCO) 3.5.1 states:

> This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10CFR50.46 (Ref. 2) will be met following a LOCA:
>
> a. Maximum fuel element cladding fuel temperature is < 2200°F;
>
> b. Maximum cladding oxidation is <0.17 times the total cladding thickness before oxidation;
>
> c. Maximum hydrogen generation from zirconium water reaction is < 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and
>
> d. Core is maintained in a coolable geometry.

While the boron concentration is based on ensuring the maintenance of a subcritical reactor in a post-LOCA environment, this also plays a part meeting the four overall goals of the LCO as stated above in that maintaining the core subcritical is an aspect of ensuring the core is maintained in a coolable geometry. By maintaining the reactor subcritical, a coolable geometry is maintained. This means that the first sentences of both C and D are correct. Further, in complying with the actions of Technical Specification 3.5.1 (as stated in Answer C), compliance with the actions of Technical Specification 3.0.3 is also maintained (making the second part of C and D also correct). Therefore, both Answers C and D are correct for the given conditions.

NRC Response for 87: The NRC agrees with the licensee's recommendation to accept both Answers C and D as correct.

.3 Initial Licensing Examination Development

a. Examination Scope

The licensee developed the examinations in accordance with NUREG-1021, Revision 9. All licensee facility training and operations staff involved in examination preparation and validation were on a security agreement. The facility licensee submitted both the written and operating examination outlines on September 12, 2007. The chief examiner reviewed the outlines against the requirements of NUREG-1021, Revision 9, and provided comments to the licensee. The facility licensee submitted the draft examination package on October 9, 2007. The chief examiner reviewed the draft examination package against the requirements of NUREG-1021, Revision 9, and provided comments to the licensee on the examination on October 22, 2007. The NRC conducted an onsite validation of the operating examinations and provided further comments during the week of October 22, 2007. The licensee satisfactorily completed comment resolution on November 15, 2007.

b. Findings

The NRC approved the initial examination outline and advised the licensee to proceed with the operating examination development.

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

No findings of significance were identified.

.4 Simulation Facility Performance

a. Examination Scope

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

b. Findings

No findings of significance were identified.

.5 Examination Security

a. Examination Scope

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

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

The chief examiner presented the examination results to Mr. David G. Lantz, Superintendent of Operations Training, on December 7, 2007. The licensee acknowledged the findings presented.

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

D. Lantz, Superintendent of Operations Training
D. Harris, ILT Supervisor
L. Wilhelm, Exam Developer

NRC Personnel

D. Dumbacher, Senior Resident Inspector

ADAMS DOCUMENTS REFERENCED

Accession No. ML080110200 – FINAL-Written Exams
Accession No. ML080110204 – FINAL-Operating Exam
Accession No. ML080110206 – FINAL-Post Exam Comments