UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
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ATLANTA, GEORGIA 30303-1257

December 23, 2010

Mr. R. M. Krich
Vice President, Nuclear Licensing
Tennessee Valley Authority
3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT – NRC OPERATOR LICENSE EXAMINATION REPORT 05000327/2010302 AND 05000328/2010302

Dear Mr. Krich:

During the period September 13-22, 2010, the Nuclear Regulatory Commission (NRC) administered operating tests to employees of your company who had applied for licenses to operate the Sequoyah Nuclear Plant. At the conclusion of the tests, the examiners discussed preliminary findings related to the operating tests with those members of your staff identified in the enclosed report. The written examination was administered by your staff on September 29, 2010.

Eight Senior Reactor Operator (SRO) applicants passed both the operating test and written examination. One SRO applicant passed a re-take written examination. Two applicants, an SRO and an RO applicant, failed only the written examination. One SRO applicant failed the operating test. There were two post-examination comments concerning the written examination and three post-examination comments concerning the operating test. The NRC's resolution to the licensee's comments is provided in this report as Enclosure 2. A Simulator Fidelity Report is included in this report as Enclosure 3.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm.adams.html (the Public Electronic Reading Room). If you have any questions concerning this letter, please contact me at (404) 562-4550.

Sincerely,

/RA/

Malcolm T. Widmann, Chief
Operations Branch
Division of Reactor Safety

Docket Nos.: 50-327, 50-328
License Nos.: DPR-77, DPR-79

Enclosures: 1. Report Details
2. NRC Resolution to Post Examination Comments
3. Simulator Fidelity Report

cc w/encl.: (See page 3)

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Letter to R. M. Krich from Malcolm T. Widmann dated December 23, 2010.

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	05000327, 05000328
License Nos.:	DPR-77, DPR-79
Report Nos.:	05000327/2010302 and 05000328/2010302
Licensee:	Tennessee Valley Authority
Facility:	Sequoyah Nuclear Plant, Units 1 and 2
Location:	Soddy-Daisy, TN
Dates:	Operating Test September 13, 2010 through September 22, 2010 Written Examination – September 29, 2010
Inspectors:	Richard S. Baldwin, Chief Examiner, Operations Engineer Robert L. Monk, Senior Resident Inspector Craig R. Kontz, Senior Project Engineer
Approved by:	Malcolm T. Widmann, Chief Operations Branch Division of Reactor Safety

SUMMARY OF FINDINGS

ER 05000327/2010302, 05000328/2010302, 09/13-22/2010 & 09/29/2010; Sequoyah Nuclear Plant; Operator License Examinations.

Nuclear Regulatory Commission (NRC) examiners conducted an initial examination in accordance with the guidelines in Revision 9, Supplement 1, of NUREG-1021, "Operator Licensing Examination Standards for Power Reactors." This examination implemented the operator licensing requirements identified in 10 CFR §55.41, §55.43, and §55.45, as applicable.

Members of Sequoyah Nuclear Plant training staff developed both the operating tests and the written examination.

The NRC administered the operating tests during the period September 13 - 22, 2010. Members of the Sequoyah Nuclear Plant training staff administered the written examination on September 29, 2010.

Eight Senior Reactor Operator (SRO) applicants passed both the operating test and written examination. One SRO applicant passed a re-take written examination. One SRO and one RO applicant failed the written examination. One SRO failed the operating test but passed the written examination. Seven SRO applicants were issued licenses. One SRO applicant received a pass letter pending certification of additional on-site time necessary to meet requirements of responsible power plant experience and one SRO applicant received a pass letter, in accordance with NUREG-1021, ES-501, Section D.3.c, Supplement 1, Rev. 9, until written examination appeals have been reviewed for impact on the licensing decision.

There were two post-examination comments on the written examination and three post-examination comments on the operating test.

No findings of significance were identified.

REPORT DETAILS

4. OTHER ACTIVITIES

4OA5 Operator Licensing Examinations

a. Inspection Scope

Members of the Sequoyah Nuclear Plant developed both the operating tests and the written examination. All examination material was developed in accordance with the guidelines contained in Revision 9, Supplement 1, of NUREG-1021, "Operator Licensing Examination Standards for Power Reactors." The NRC examination team reviewed the proposed examination. Examination changes agreed upon between the NRC and the licensee were made per NUREG-1021 and incorporated into the final version of the examination materials.

The NRC reviewed the licensee’s examination security measures while preparing and administering the examinations in order to ensure compliance with 10 CFR Part 55.49, “Integrity of examinations and tests.”

The NRC examiners evaluated one Reactor Operator (RO) and ten Senior Reactor Operator (SRO) applicants using the guidelines contained in NUREG-1021. The examiners administered the operating tests during the period September 13, 2010, through September 22, 2010. Members of the Sequoyah Nuclear Plant training staff administered the written examination on September 29, 2010. Evaluations of applicants and reviews of associated documentation were performed to determine if the applicants, who applied for licenses to operate the Sequoyah Nuclear Plant, met the requirements specified in 10 CFR Part 55, “Operators’ Licenses.”

b. Findings

No findings of significance were identified.

Nine SRO applicants passed both the operating test and written examination. One SRO applicant passed the written examination but did not pass the operating test. One SRO and one RO failed the written examination but passed the operating test. Seven SRO applicants were issued licenses.

One SRO applicant passed the operating test and the written examination but was not issued a license until the facility licensee has certified that the required on-site time has been completed. One SRO applicant passed the operating test, but passed the SRO-only portion of the written examination with a score between 70% and 74%. Each of these applicants was issued a letter stating that they passed the examination and issuance of their license has been delayed pending the completion of the on-site time requirement for one SRO requiring that and for one SRO who received a 72% and any written examination appeals that may impact the licensing decision for their application.

Copies of all individual examination reports were sent to the facility Training Manager for evaluation of weaknesses and determination of appropriate remedial training.

The licensee submitted two post-examination written comments and three post-examination operating test comments, a copy of these may be accessed in the ADAMS system (ADAMS Accession Numbers ML103000225 and ML1034405541). A copy of the final RO and SRO written examinations, handouts and answer keys, with all changes incorporated may be accessed in the ADAMS system (ADAMS Accession Numbers ML103000228).

4OA6 Meetings, Including Exit

Exit Meeting Summary

On September 22, 2010, the NRC examination team discussed generic issues associated with the operating test with Mr. Chris Church, Sequoyah Nuclear Plant Site Vice President, and members of the Sequoyah Nuclear Plant staff. The examiners asked the licensee if any of the examination material was proprietary. No proprietary information was identified.

KEY POINTS OF CONTACT

Licensee personnel

C. Church, Site Vice President
K. Langdon, Plant Manager
D. Hawes, Operations Training Supervisor
A. Bergeron, Operations Training Manager
B. Wetzel, Site Licensing Manager
S. Smith, Operations Training
P. Simmons, Operations Manager
C. Ware, Training Director

NRC personnel

C. Young, SRI

NRC Resolution to Post-Examination Comments

A complete Text of the licensee's post examination comments can be found in ADAMS under Accession Number ML103000225 and ML1034405541.

RO QUESTION # 41

LICENSEE COMMENT:

The first part of the question concerns itself with determination of a given temperature and the ability to identify if it was in between the "optimal" temperature range for operation of the ice condenser in accordance with procedure 0-SO-61-1, "Ice Condenser Cooling." The second part of the question concerns itself with identifying the adverse affect if the temperature was outside of **this** "optimal" temperature range.

The licensee contended, in their post-examination comment, that the question has two correct answers, the original answer "A" and an additional correct answer "B." The licensee pointed out that the optimal temperature was 18° to 20° Fahrenheit (F). Both answers "A" and "B" have the same correct temperature, 19°F. The licensee pointed out, that one applicant was not sure how far out of the optimal range the question was inferring, so the applicant "assumed" that the temperature was greater than 27°F, the Technical Specification (TS) Limit. When greater than the TS limit the applicant could then justify the answer in the second part of distractor B, (operation outside Technical Specification limit). This assumption allowed the applicant to determine that the TS were not met and entry into a Limiting Condition for Operation (LCO) had to be entered and concluded that the plant may have to shutdown.

The licensee referred to procedure 1-SO-OPS-000-002.0, "Daily Shift Log," 3.0 Precautions and Limitations, C. This reference states; "The Ice Condenser may be operated at a temperature outside the optimum range, provided technical specifications limits are maintained, in accordance with Engineering recommendations to support system or planned outages." The licensee then stated that "...the plant could operate outside the optimal limit as long as the Technical Specification limit was not exceeded." They contend that this would make "B" also a correct answer.

The licensee stated that the intent of the question's answer was to determine "adverse affects" outside the optimal range that were related to the physical changes. The licensee further stated that the Final Safety Analysis Report (FSAR) 6.5.7.2 identified optimal temperature in order to minimize physical changes. They contend that one could assume the question did not limit itself to just physical changes but also to administrative actions (operation outside of Technical Specifications). Thus making both "A" and "B" correct.

NRC Resolution to Post-Examination Comments

NRC RESOLUTION:

The NRC disagrees with this comment.

When the applicant assumed something to answer the question, the applicant then placed different parameters on the question. When assumptions were used to answer a question, the question then became something it was not intended to be. As is stated above in the argument why answer “B” should be allowed, the licensee refers to a procedure, 1-SO-OPS-000-002.0, “Daily Shift Log,” 3.0 Precautions and Limitations, “C.” They identify in this Precaution and Limitation that the plant is allowed to operate outside the “optimal limit as long as the technical specification limit was not exceeded.” In the licensee’s discussion it is stated that the applicant had to make an assumption that the TS limit was exceeded. This is contrary to Precaution and Limitation, “C.” Additionally, during the NRC briefing for the written examination, Appendix E, “Policies and Guidelines for Taking NRC Examinations,” of NUREG 1021 was used to brief the applicants prior to administration of the written examination. Specifically, Part B, “Written Examination Guidelines,” B.7 was read to all the applicants. B.7 states in part “…When answering a question, do *not* make assumptions regarding conditions that are not specified in the question…”

Based on the above discussion, the answer to this question was limited to the original answer “A.” No change to the answer key was warranted.

SRO QUESTION # 95

LICENSEE COMMENT:

The question concerns itself with and outage on Unit 1 and the use of schedule logic changes. The question proposed four answers that were to be analyzed using procedure, SPP-7.2, “Outage Management,” Appendix E, “Outage Schedule Logic Change Control,” for determination of logic changes that would have met the criteria for a Safety Significant change.

The licensee contended that there were two answers to this question, “A” and “D.” The licensee contended that since the Condenser Circulating Water System (CCW) intake bay was common to both units, diving operations could affect both units. The licensee additionally stated that when evaluating safety significance the SRO used SPP 7.2, Appendix E to determine if the activity would meet any of the listed criteria. The licensee pointed out that Item 15 asked if it will “affect the non-outage unit?” The licensee identified that diving in the vicinity of the CCW pumps and with industry OE (none provided) and that it could (would) affect the non-outage unit, the licensee considered both answers “A” and “D” as correct answers.

NRC RESOLUTION:

The NRC disagrees with this comment.

The question asks the applicant to act as an SRO to determine if any of the distractors identified a Safety Significant Change and to do this using procedure, SPP-7.2. While the NRC agrees

NRC Resolution to Post-Examination Comments

that diving operations affect both units, however, in this case for distractor "D" to qualify as an additional correct answer, one must view the distractor in its entirety. The distractor provided amplifying information concerning the operation of the CCW pumps. The distractor stated when the diving operations are occurring. Initially diving operations are scheduled at the end of the outage and they would be done before starting the CCW pumps. When moving it to the beginning of the outage, diving operations are scheduled after securing the CCW pumps. The safety significance of diving operations was not contested, just when these operations were accomplished, at the beginning or the end of the outage. It is the NRC's view that there was no change in safety significance of this operation and therefore doing the action at the beginning or the end of the outage would make no difference because the CCW system was in exactly the same alignment. Since this was an allowable evolution, the conditions provided are the same.

Based on the above discussion, the answer to this question was limited to the original answer "A." No change to the answer key is warranted.

ADMINISTRATIVE JPM SRO D: CLASSIFY THE EVENT per the REP (SGTR with Failed S/G Safety)

LICENSEE COMMENT:

The JPM concerns itself with classification of an event using the licensee's Emergency Action Levels (EALs). Based on the information provided, the applicant was expected to base the EAL classification on the loss of three of the fission product barriers, which required the applicant to classify the event as a General Emergency (GE). Additionally, the applicants were provided significant information that allowed the determination of a Steam Generator Tube Rupture (S/GTR).

The licensee provided justification as to why specific lines in the JPM were considered or *not* considered critical. The licensee indicated that Appendix C, TVA Initial Notification of General Emergency," Line 4, was initially considered a critical step and provided justification as to why the step should *not* be critical. They pointed out that the EAL number was the critical part because outside agencies used this information to correctly determine events in progress. The description was helpful. However, it was not used to identify the event(s) that are taking place. Therefore the licensee recommended that Line 4 be considered as a non-critical step and therefore, not constitute a failure of the JPM if done incorrectly.

For Line 5, Radiological Conditions, the licensee identified two answers, the initial answer "Release Information not known," and an additional answer "Releases above federally approved limits." The licensee stated that these two determinations should have been used for both Airborne, as well as, Liquid releases Offsite. The licensee did not provide any justification for the additional answer. The initial conditions provided, only concerns itself with a Steam Generator Tube Rupture and an airborne release. Nothing in the initial conditions would lead the applicant to determine that a liquid release was or had occurred.

NRC Resolution to Post-Examination Comments

For Line 7, “Meteorological Conditions, the initial JPM key identified that the wind direction was identified as the “instantaneous” wind speed and should have been the 15 minute average. The wind direction, which was critical for determining the sectors for evacuation was correct. Additionally, the Wind Speed” was initially marked as a critical portion of the JPM. Thus, the correct answer for Line 7 was to use the 46 meter height wind direction (a critical step). The wind direction was for information only and had no critical element for JPM completion.

NRC RESOLUTION:

The NRC partially agreed with the licensee’s comment.

The critical part (critical step) of Line 4, was pointed out by the facility licensee, was the correct EAL accident designator. The NRC agreed that this determination was critical and the specific words used by an applicant could have been interpreted different ways. The NRC agrees that the only part of Line 4 critical to accomplish the JPM correctly was the exact EAL designator. The JPM was changed to reflect this correction.

For Line 5, the NRC partially agrees with the facility licensee’s assessment. The initial conditions of this JPM clearly identified that an airborne leak was occurring and based on that, the information was not conclusive that the Release Rate would be above federally approved limits. Because of the uncertainty of the released amount, the answer “Releases above federally approved limits,” was not accepted as an additional correct answer. Also, since the initial conditions did not specifically state there was a liquid release, the NRC did not agree that this was an additional answer. The JPM was not changed and the only correct answer considered for Line 5 was under “Airborne release” and identified as “Release Information not known.”

For Line 7, the licensee pointed out that it was incorrect to use the 15 minute instantaneous value for wind direction and identified that it should have been the 15 minute average value. The NRC agrees with this comment and the answer key was changed to reflect this change. Additionally, the facility licensee identified that initially the direction from which the wind was from was considered a critical step and that the speed of that wind was not critical to determine protective measures. The NRC agreed with this comment. Line 7 was changed to reflect the 15 minute “average” wind direction measured from 46 meters was considered critical and that the wind speed would not be critical to the JPM.

CONTROL ROOM – SYSTEM JPM D: Initiate Makeup to the Refueling Cavity

LICENSEE COMMENT:

This alternate path JPM concerned itself with the applicant having to determine that the procedure flow path from the body of the procedure to Appendix A, “Filling Refueling Cavity from the RWST,” could not be accomplished due to a failure of valves FCV-62-135, and FCV-62-136, “CCP Suction from RWST,” to open. The applicant was expected to continue on in Appendix A and perform Step 2, “Initiating Make-up from the RWST using RHR pump suction.”

NRC Resolution to Post-Examination Comments

The licensee identified that initially the JPM was designed to use Appendix A to fill the Refueling Cavity using the Centrifugal Charging Pumps (CCPs). During the performance of the procedure, the applicant identified that the CCPs were not available for fill, and then the applicant was expected to proceed on in Appendix A and align the RHR (Residual Heat Removal) pumps by taking suction from the RWST (Step 2). The licensee pointed out that, if, Step 1 of the appendix was not able to be completed/performed then it was "reasonable" for the applicant to return to the main body of procedure AOP-M.04, "Refueling Malfunctions." When the applicant returned to Step 3.a, the applicant should then "**INITIATE** makeup from RWST **USING** Appendix "A," Fill Refueling Cavity from RWST." Since this step was not able to be performed, the applicant was expected to go to the Response Not Obtained (RNO) column of Step 3.a. The RNO directed the applicant to perform Appendix. E, "Refueling Cavity Makeup Using Normal Charging." Additionally, the licensee pointed out that the initial conditions provided direction to establish makeup from the RWST; however, using normal charging line-up in Appendix E was also a logical method.

The licensee further discussed flow rates when the CCPs have suction from the VCT (~120 gpm), the RWST (~400 gpm) and finally the RHR pumps with suction from the RWST (Step 2 of Appendix A), (~3000 gpm). The directions in the JPM did not direct nor indicate the volume of water or flow rate necessary, it only required filling the refueling cavity.

NRC RESOLUTION:

The NRC agreed with the licensee's comment.

The JPM directed the applicant to use AOP-M.04, to mitigate the event being presented on the simulator. The applicant did not know that the design of the JPM was to require a decision to fill the refueling cavity using the RHR pumps. Attachment "A" of AOP-M.04 was not written in the two column format as was the main body of the procedure. The NRC agreed with the facility licensee that because it was not in the two column format it was reasonable for an operator to return to the main body of the procedure when not being able to accomplish Attachment "A." When going back to Step 3 of AOP-M.04, the applicant could perform the RNO of that step or procedurally from the RNO, go to Appendix "E" and use the CCPs, with suction from the VCT.

The licensee correctly pointed out that there was no information as to the severity of the leak out of the canal. Since this information was not provided, it did not matter from what source the canal was refilled, only that it was being refilled. The RNO directed the applicant to use Attachment "E" of AOP-M.04 and use the CCPs with suction from the VCT.

The JPM was changed to allow either Appendix "A" or Appendix "E" of AOP-M.04 correct methods for filling the refueling canal.

ADMINISTRATIIVE TOPIC, JPM A.1.A, Overtime Restrictions.

The JPM concerns itself with an in depth schedule of a licensed operator and required the applicant to determine dates of violation(s), if any, of the Fatigue Rule that would require an

NRC Resolution to Post-Examination Comments

"Overtime Exception Report" to be completed prior to returning to work. It was expected that all violation(s) be explained with the reason for the violation. There were two instances where the individual exceeded 26 hrs in a 48 hr period. There was a question concerning the use of the rule that there must be "At least a 34 hour break in any 9 calendar day period." This concern was identified to the facility licensee during the examination week.

The licensee pointed out that there is a software program used at the site for Personnel Qualification and Scheduling in the Shift Operations Management System (eSoms). This program takes into account all the requirements found in NPP-SPP-03.21, "Fatigue Management and Work Hour Limits," and was used to ensure actual or projected hours at the site were in compliance with 10CFR26, "Fitness for Duty," requirements. The licensee used this system to determine if there were any other answers for this JPM. There was a question concerning the requirement of Section 3.2.1, "10 Code of Federal Regulation (CFR) 26 Overtime Limits," A.5. "At least a 34 hour break in any 9 calendar day period." When the JPM's data was input into the eSoms program it was found that Section 3.2.1.A.5 requirements were not in violation. The licensee recommended that if an applicant identified this as an additional potential violation, over and above the two actual original violations, that they be considered, in this instance, to be a conservative call.

NRC RESOLUTION:

10CFR26.205 (d)(2)(ii) Work Hour Controls: Requires "A 34-hour break in any 9-day period."

HQ has further clarified the requirement on the FAQ page as follows:
http://www.nrc.gov/reactors/operating/ops-experience/fitness-for-duty-programs/faqs/manage-fatigue.html

This is a excerpt from the FAQ page:

To implement the required 34-hour break for each 9-day period is it necessary to use a "sliding window" to ensure that every single 216-hour period contains a complete 34-hour break.

"The licensee must conduct two types of assessments to ensure that each individual who is subject to the requirements of Subpart I, receives at least a 34-hour break during any 9-day period per § 26.205(d)(2)(ii). The first assessment addressed shift scheduling and the second assessment addressed actual hours worked.

To comply with the scheduling requirements in § 26.205(c), the licensee must verify that at the start of each shift, the individual is scheduled to have a 34-hour break within the following 216 hours.

To ensure that the actual hours worked by the individual are in compliance with the requirements of Subpart I, the licensee must confirm that at the end of the individual's forthcoming work period, including overtime if applicable – the individual has had a 34-hour break within the preceding 216 hours (i.e., nine 24-hour periods, rather than calendar days). The licensee must perform this check before allowing an individual to work any given period, scheduled or otherwise. Licensees should be particularly watchful with regard to allowing

NRC Resolution to Post-Examination Comments

overtime immediately prior to a scheduled break of 34 hours or more to ensure that the resulting break complies with the provisions in Subpart I.

The licensee must continuously look forward, in real time, from the start of the first period of work, immediately following a 34 hour break, to ensure there is a 34 hour break in the subsequent 216 hour period."

It is not necessary to check all other possible 216-hour windows.

By the wording in **NPP-SPP-03.21**, it was actually more restrictive than was required by the NRC. However, it was not being implemented in that fashion at the site as verified in eSOMS. Therefore, the JPM required the applicants to assess the work hours using the site procedure, and the procedure was not explicit in the application of the "34 Hour Break in any 9 Day Period" requirement, the NRC finds it acceptable that the identification of the "34 hour Break…" was not considered failure criteria, but do consider this to be level of knowledge weakness in those applicants that indicated this was an issue with the schedule provided for analysis.

Simulator Fidelity Report

Facility Licensee: Sequoyah Nuclear Plant

Facility Docket No.: 05000327, 05000328

Operating Test Administered: September 13 – 22, 2010

This form is to be used only to report observations. These observations do not constitute audit or inspection findings and, without further verification and review in accordance with Inspection Procedure 71111.11 are not indicative of noncompliance with 10 CFR 55.46. No licensee action is required in response to these observations.

The following simulator fidelity or configuration issues were identified:

NONE

system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm.adams.html (the Public Electronic Reading Room). If you have any questions concerning this letter, please contact me at (404) 562-4550.

Sincerely,

/RA/

Malcolm T. Widmann, Chief
Operations Branch
Division of Reactor Safety

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cc w/encl.: (See page 3)

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